UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 11, 2017

Mr. G. T. Powell
Executive Vice President and CNO
STP Nuclear Operating Company
South Texas Project
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: CHANGES TO DESIGN BASIS ACCIDENT ANALYSIS USING A RISK-INFORMED METHODOLOGY TO ACCOUNT FOR DEBRIS IN CONTAINMENT (CAC NOS. MF2400 AND MF2401)

Dear Mr. Powell:

By letter dated June 19, 2013, as supplemented by letters dated October 3, October 31, November 13, November 21, and December 23, 2013 (two letters); January 9, February 13, February 27, March 17, March 18, May 15 (two letters), May 22, June 25, and July 15, 2014; March 10, March 25, and August 20, 2015; April 13, May 11, June 9, June 16, July 18, July 21 (two letters), July 28, September 12, October 20, and December 7, 2016; and January 19, 2017, STP Nuclear Operating Company (STPNOC, the licensee) submitted a license amendment application and associated exemption requests for South Texas Project (STP), Units 1 and 2, as the pilot plant for a risk-informed resolution to Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR [pressurized-water reactor] Sump Performance," and to close Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004.

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 212 to Facility Operating License No. NPF-76 and Amendment No. 198 to Facility Operating License No. NPF-80 for STP, Units 1 and 2, respectively. The license amendments authorize revision of the design-basis accident analysis described in the Updated Final Safety Analysis Report and revise the STP Technical Specifications for the emergency core cooling system (ECCS) and containment spray system (CSS) specific to the impacts of debris in containment. The NRC staff's safety evaluation (SE) is provided in Enclosure 3.

The combined risk-informed and deterministic analysis methodology termed Risk over Deterministic, or RoverD, discussed in Section 1.3 of the enclosed SE, was a unique feature of this pilot proposal and received significant scrutiny. The NRC staff met with STPNOC staff and contractors in over 40 public meetings since 2011. The NRC staff observed plant-specific testing conducted at Texas A&M University and at Alden Laboratories in Holden, Massachusetts, to evaluate the STP's sump strainers response to debris. The NRC staff participated in 13 audits in Washington, DC; College Station, Texas; Holden, Massachusetts; and Albuquerque, New Mexico, contractor sites. The NRC staff audits included entry into the STP containment to assess model accuracy, flow paths, and insulation and coatings condition. The licensee responded to over 400 questions posed in NRC staff requests for additional information. The NRC staff also adopted recommendations from the Advisory Committee on Reactor Safeguards (ACRS) following meetings in September 2014, March 2015, April 2017, and May 2017. The ACRS issued a letter on May 17, 2017 (ADAMS Accession No. ML17137A325), to the Commission concluding that STPNOC's proposed changes to its licensing basis and technical specifications are acceptable.

The NRC staff assessed the results of STPNOC's implementation of the RoverD assessment methodology on the STP containment sumps performance considering the effects of debris as described in Enclosure 3. The NRC staff reviewed your submittal using the five key principles of risk-informed regulation specified in Regulatory Guide 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," May 2011 (ADAMS Accession No. ML100910006), and other guidance, to prepare a comprehensive SE. The NRC staff concluded that the five key principles of risk-informed regulation have been met, and, thus, the RoverD methodology and results provide reasonable assurance that the ECCS and CSS structures, systems, and components will remain capable of performing their safety-related functions, considering the effects of debris, without endangering the health and safety of the public.

Attachment 2 of the SE documents the NRC staff's review of your use of RELAP5-3D as the platform to evaluate in-core thermal-hydraulic effects of debris on long-term core cooling following a loss-of-coolant accident (LOCA). The NRC staff reviewed the RELAP5-3D platform specifically for STPNOC's use as a proposed long-term core cooling application since it is not an NRC-approved code for this application. The NRC staff concluded that the STPNOC long-term core cooling evaluation model is an acceptable evaluation model for debris impacts for hot-leg breaks 16 inches in diameter and smaller. Further, the simulations performed with this evaluation model along with those from the LOCA Disposition Model demonstrate that the acceptance criteria have been satisfied. The findings in the NRC staff's SE are limited to STPNOC and are not a generic approval of the RELAP5-3D platform. As specified in SE Attachment 2 and SE Section 7.0, RELAP5-3D is acceptable only for use by STPNOC for specific application at STP and as limited by the SE. Prior NRC review and approval is needed for other licensees to use RELAP5-3D for similar applications.

On February 8 and 17, 2017, we requested that you review a preliminary SE for factual errors or omissions. You provided your comments on February 14 and March 1, 2017. Enclosure 4 provides our response to your comments as discussed with your staff.

As part of the license amendment application, STPNOC requested exemptions from the requirements of Title 10 to the *Code of Federal Regulations* (10 CFR), Section 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR 50, Appendix A, General Design Criterion (GDC) 35, "Emergency core cooling," GDC 38, "Containment heat removal," and GDC 41, "Containment atmosphere clean-up." The exemption requests were necessary since the NRC has interpreted these regulations as requiring a deterministic approach and bounding calculation to show compliance with ECCS and CSS requirements. The NRC staff relied on its SE as part of the basis for granting the exemptions under 10 CFR 50.12. The exemption requests are discussed in a separate letter (ADAMS Accession No. ML17037C871), and in a *Federal Register* notice to be published shortly.

The Commission supported the option of a risk-informed method to resolve GSI-191 in a December 14, 2012, Staff Requirements Memorandum associated with SECY-12-0093, "Closure Options for Generic Safety Issue 191, Assessment of Debris Accumulation on

Pressurized-Water Reactor Sump Performance" (ADAMS Accession No. ML12349A378). The NRC staff has prepared a draft final rule, 10 CFR 50.46c, which, if promulgated, would allow a risk-informed approach to account for debris without the need for exemptions.

The NRC staff published an Environmental Assessment on the exemption requests in the *Federal Register* on May 9, 2017 (82 FR 21568), finding that the exemptions will not have a significant impact on the environment. A draft Environmental Assessment was published in the *Federal Register* for comments on May 4, 2016 (81 FR 26838). No public comments were received.

A Notice of Issuance of the amendments will be included in the Commission's next biweekly *Federal Register* notice. If you have any questions, please do not hesitate to contact me at 301-415-1906 or Lisa.Regner@nrc.gov.

Sincerely,

Lisa M. Regner, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosures:
1. Amendment No. 212 to NPF-76
2. Amendment No. 198 to NPF-80
3. Safety Evaluation
4. Resolution of Licensee Comments on NRC Staff Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 212
License No. NPF-76

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by STP Nuclear Operating Company (STPNOC)*, acting on behalf of itself and for NRG South Texas LP, the City Public Service Board of San Antonio (CPS), and the City of Austin, Texas (COA) (the licensees), dated June 19, 2013, as supplemented by letters dated October 3, October 31, November 13, November 21, and December 23, 2013 (two letters); January 9, February 13, February 27, March 17, March 18, May 15 (two letters), May 22, June 25, and July 15, 2014; March 10, March 25, and August 20, 2015; April 13, May 11, June 9, June 16, July 18, July 21 (two letters), July 28, September 12, October 20, and December 7, 2016; and January 19, 2017, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

*STPNOC is authorized to act for NRG South Texas LP, the City Public Service Board of San Antonio, and the City of Austin, Texas, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, by Amendment No. 212, the license is amended to authorize revisions to the Updated Final Safety Analysis Report (UFSAR), as set forth in the applications dated August 20, 2015, and October 20, 2016. The licensee shall update the UFSAR to incorporate Attachment 3-4 as described in the licensee's application dated August 20, 2015, and Attachment 3-4 of the licensee's application dated October 20, 2016, and the NRC staff's safety evaluation associated with this amendment, and shall submit the revised description authorized by this amendment with the next update of the UFSAR.

3. Additionally, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-76 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 212, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

4. The license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance. The UFSAR changes shall be implemented in the next periodic update to the UFSAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License No. NPF-76 and the Technical Specifications

Date of Issuance: July 11, 2017

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-499

SOUTH TEXAS PROJECT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 198
License No. NPF-80

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by STP Nuclear Operating Company (STPNOC)*, acting on behalf of itself and for NRG South Texas LP, the City Public Service Board of San Antonio (CPS), and the City of Austin, Texas (COA) (the licensees), dated June 19, 2013, as supplemented by letters dated October 3, October 31, November 13, November 21, and December 23, 2013 (two letters); January 9, February 13, February 27, March 17, March 18, May 15 (two letters), May 22, June 25, and July 15, 2014; March 10, March 25, and August 20, 2015; April 13, May 11, June 9, June 16, July 18, July 21 (two letters), July 28, September 12, October 20, and December 7, 2016; and January 19, 2017, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

*STPNOC is authorized to act for NRG South Texas LP, the City Public Service Board of San Antonio, and the City of Austin, Texas, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, by Amendment No. 198, the license is amended to authorize revisions to the Updated Final Safety Analysis Report (UFSAR), as set forth in the applications dated August 20, 2015, and October 20, 2016. The licensee shall update the UFSAR to incorporate Attachment 3-4 as described in the licensee's application dated August 20, 2015, and Attachment 3-4 of the licensee's application dated October 20, 2016, and the NRC staff's safety evaluation associated with this amendment, and shall submit the revised description authorized by this amendment with the next update of the UFSAR.

3. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-80 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 198, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

4. The license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance. The UFSAR changes shall be implemented in the next periodic update to the UFSAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License No. NPF-80 and the Technical Specifications

Date of Issuance: July 11, 2017

ATTACHMENT TO LICENSE AMENDMENT NOS. 212 AND 198 TO

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

Replace the following pages of the Facility Operating Licenses, Nos. NPF-76 and NPF-80, and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License No. NPF-76

REMOVE	INSERT
-4-	-4-

Facility Operating License No. NPF-80

REMOVE	INSERT
-4-	-4-

Technical Specifications

REMOVE	INSERT
3/4 5-3	3/4 5-3
---	3/4 5-3a
3/4 6-14	3/4 6-14
---	3/4 6-14a

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 212, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Not Used

(4) Initial Startup Test Program (Section 14, SER)*

Any changes to the Initial Test Program described in Section 14 of the Final Safety Analysis Report made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Safety Parameter Display System (Section 18, SSER No. 4)*

Before startup after the first refueling outage, HL&P[**] shall perform the necessary activities, provide acceptable responses, and implement all proposed corrective actions related to issues as described in Section 18.2 of SER Supplement 4.

(6) Supplementary Containment Purge Isolation (Section 11.5, SSER No. 4)

HL&P shall provide, prior to startup from the first refueling outage, control room indication of the normal and supplemental containment purge sample line isolation valve position.

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* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

** The original licensee authorized to possess, use and operate the facility was HL&P. Consequently, historical references to certain obligations of HL&P remain in the license conditions.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 198 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Not Used

(4) Initial Startup Test Program (Section 14, SR)*

Any changes to the Initial Test Program described in Section 14 of the Final Safety Analysis Report made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) License Transfer

Texas Genco, LP shall provide decommissioning funding assurance, to be held in decommissioning trusts for South Texas Project, Unit 2 (Unit 2) upon the direct transfer of the Unit 2 license to Texas Genco, LP, in an amount equal to or greater than the balance in the Unit 2 decommissioning trust immediately prior to the transfer. In addition, Texas Genco, LP shall ensure that all contractual arrangements referred to in the application for approval of the transfer of the Unit 2 license to Texas Genco, LP to obtain necessary decommissioning funds for Unit 2 through a non-bypassable charge are executed and will be maintained until the decommissioning trusts are fully funded, or shall ensure that other mechanisms that provide equivalent assurance of decommissioning funding in accordance with the Commission's regulations are maintained.

(6) License Transfer

The master decommissioning trust agreement for Unit 2, at the time the direct transfer of Unit 2 to Texas Genco, LP is effected and thereafter, is subject to the following:

* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

3/4.5.2 ECCS SUBSYSTEMS - T_{AVG} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Three independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE High Head Safety Injection pump,

b. One OPERABLE Low Head Safety Injection pump,

c. One OPERABLE RHR heat exchanger, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation through a High Head Safety Injection pump and into the Reactor Coolant System and through a Low Head Safety Injection pump and its respective RHR heat exchanger into the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

a. With less than the above subsystems OPERABLE, but with at least two High Head Safety Injection pumps in an OPERABLE status, two Low Head Safety Injection pumps and associated RHR heat exchangers in an OPERABLE status, and sufficient flow paths to accommodate these OPERABLE Safety Injection pumps and RHR heat exchangers,** within 7 days restore the inoperable subsystem(s) to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With less than two of the required subsystems OPERABLE, within 1 hour restore at least two subsystems to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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* Entry into MODE 3 is permitted for the Safety Injection pumps declared inoperable pursuant to Specification 4.5.3.1.2 provided that the Safety Injection pumps are restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

** Verify required pumps, heat exchangers and flow paths OPERABLE every 48 hours.

3/4.5.2 ECCS SUBSYSTEMS - T_{AVG} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

c. With less than the required flow paths OPERABLE solely due to potential effects of LOCA generated and transported debris that exceeds analyzed amounts, perform the following:

   1. immediately initiate action to implement compensatory actions,

      AND

   2. within 90 days restore the affected flowpath(s) to OPERABLE status,

      OR

   Be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be submitted within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Three independent Containment Spray Systems shall be OPERABLE with each Spray system capable of taking suction from the RWST and transferring suction to the containment sump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one Containment Spray System inoperable, within 7 days restore the inoperable Spray System to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

b. With more than one Containment Spray System inoperable, within 1 hour restore at least two Spray Systems to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one or more Containment Spray Systems inoperable in MODE 1, 2, or 3 solely due to potential effects of LOCA generated and transported debris that exceeds analyzed amounts, perform the following:

   1. immediately initiate action to implement compensatory actions,

   AND

   2. within 90 days restore the affected system(s) to OPERABLE status,

   OR

   Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. By verifying on a STAGGERED TEST BASIS, that on recirculation flow, each pump develops a differential pressure of greater than or equal to 283 psid when tested pursuant to Specification 4.0.5;

c. At a frequency in accordance with the Surveillance Frequency Control Program during shutdown, by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure High 3 test signal, and

   2) Verifying that each spray pump starts automatically on a Containment Pressure High 3 test signal coincident with a sequencer start signal.

d. By verifying each spray nozzle is unobstructed following maintenance activities that could result in spray nozzle blockage.

ENCLOSURE 3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 212 AND 198 TO

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

STP NUCLEAR OPERATING COMPANY, ET AL.

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

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ATTACHMENTS:

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 212 AND 198 TO

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

STP NUCLEAR OPERATING COMPANY, ET AL.

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

1.0 INTRODUCTION

1.1 Application

By application dated June 19, 2013,[1] STP Nuclear Operating Company (STPNOC, the licensee) submitted license amendment requests (LARs), as supplemented by letters dated October 3,[2] October 31,[3] November 13,[4] November 21,[5] and December 23, 2013 (two letters);[6,7] January 9,[8] February 13,[9] February 27,[10] March 17,[11] March 18,[12] May 15 (two letters),[13,14] May 22,[15] June 25,[16] and July 15, 2014;[17] March 10,[18] March 25,[19] and August 20, 2015;[20] and April 13,[21] May 11,[22] June 9,[23] June 16,[24] July 18,[25] July 21 (two letters),[26,27] July 28,[28] September 12,[29] October 20,[30] November 9,[31] and December 7, 2016;[32] and January 19, 2017,[33] as the pilot plant to adopt a risk-informed resolution to Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance," as its licensing basis analysis, and to close U.S. Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004,[34] for the South Texas Project, Units 1 and 2 (STP).

The licensee submitted these requests to use a risk-informed methodology as allowed by the Commission in a Staff Requirements Memorandum (SRM) associated with SECY-12-0093, "Closure Options for Generic Safety Issue-191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," dated December 14, 2012.[35] The amendments would authorize Updated Final Safety Analysis Report (UFSAR) revisions for STPNOC to use a risk-informed approach to resolve the NRC staff concerns stated in GSI-191 and GL 2004-02. The licensee also requested that the STP operating licenses be amended to revise Technical Specification (TS) 3/4.5.2, "ECCS [Emergency Core Cooling System] Subsystems – Tavg Greater Than or Equal to 350°F," and TS 3/4.6.2, "Depressurization and Cooling Systems, Containment Spray System," to add a required action and completion time specific to the effects of debris.

Portions of the letters dated January 9, February 13, and March 18, 2014, and May 11, 2016, contain sensitive unclassified non-safeguards information and have been withheld from public disclosure pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations* (10 CFR).

The January 9 and March 18, 2014, submittal attachments are withheld in their entirety since they are RELAP5-3D code information. The February 13, 2014, and May 11, 2016, documents are available in non-proprietary versions.

1.2 Background

1.2.1 Challenges to Safety Systems Function from Debris in Containment

The function of the emergency core cooling system (ECCS) is to cool the reactor core and provide shutdown capability following a loss-of-coolant accident (LOCA). The primary functions of the containment spray system (CSS) are to reduce containment pressure and to reduce the concentration and quantity of fission products in the containment building after a LOCA.

Nuclear plants are designed and licensed with the expectation that they are able to remove reactor decay heat following a LOCA to prevent core damage. Long-term cooling following a LOCA is also a basic safety function for nuclear reactors. The recirculation sump located in the lower areas of the reactor containment structure provides a water source to the ECCS in a pressurized-water reactor (PWR) once the initial water source has been depleted and the systems are switched over to recirculation mode for extended cooling of the core. For a more detailed explanation of these systems, refer to Section 2.1, "Description of Affected Structures, Systems, and Components," of this safety evaluation (SE).

If a LOCA occurs, piping thermal insulation and other materials located in containment may be dislodged by the two-phase (steam and liquid) water jet emanating from the broken pipe. This debris may be transported by the flow of water and steam from the break or from the CSS, to the pool of water that collects in the containment recirculation sump. Once transported to the sump pool, the debris could be drawn towards the ECCS sump strainers, which are designed to prevent debris from entering the CSS and the ECCS. If this debris clogged the strainers, the ECCS could fail, resulting in core damage, or the CSS pumps could fail, resulting in containment pressure or radiation dose increasing beyond deterministic limits. It is also possible that some debris could bypass the sump strainers and lodge in the reactor core. This could result in reduced core cooling and potential core damage.

Findings from research and industry operating experience raised questions concerning the adequacy of PWR sump designs. Research findings demonstrated that the amount of debris generated and transported by a high-energy LOCA could be greater than originally anticipated. The debris from a LOCA could also be finer, and thus more easily transportable, and could be comprised of debris consisting of fibrous material combined with particulate material that could result in a substantially greater flow restriction than an equivalent amount of either type of debris alone. These research findings prompted the NRC to open a generic safety issue.

1.2.2 Generic Safety Issue 191

In 1996, the NRC identified GSI-191 associated with the effects of debris accumulation on PWR sump performance during design-basis accidents (DBAs). This was identified from similar concerns at boiling-water reactors, new information identified following closure of the actions taken for resolution of the GSI at boiling-water reactors, and confirmatory testing conducted by the NRC.

Resolution of GSI-191 involves two distinct but related safety concerns: (1) potential clogging of the sump strainers that results in ECCS and/or CSS pump failure, and (2) potential clogging of flow channels within the reactor vessel because of debris bypassing the sump strainers, often referred to as in-vessel effects. Clogging at either the strainers or in-vessel channels can result in loss of the long-term cooling safety function.

Significant progress has been made by the NRC, licensees, Nuclear Energy Institute (NEI), Pressurized Water Reactor Owners Group (PWROG), and other external stakeholders to resolve the concerns defined by GSI-191. More information on the background, testing, and other actions associated with GSI-191 can be found in NUREG-0897, "Containment Emergency Sump Performance: Technical Findings Related to Unresolved Safety Issue A-43," October 1985,[36] and NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors," dated June 9, 2003.[37]

1.2.3 Generic Letter 2004-02

As part of the actions to resolve GSI-191, in September 2004, the NRC issued GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors," to holders of operating licenses for PWRs. In GL 2004-02, the NRC staff requested that licensees perform an evaluation of their ECCS and CSS recirculation functions considering the potential for debris-laden coolant to be circulated by the ECCS and the CSS after a LOCA or high-energy line break (HELB) inside containment and, if appropriate, take additional action to ensure system function. The GL 2004-02 required, per 10 CFR 50.54(f), that licensees provide the NRC a written response describing the results of their evaluation and any modifications made, or planned, to ensure ECCS and CSS system function during recirculation following a design basis event, or any alternate action proposed and the basis for its acceptability. For more specifics of the required information to respond to GL 2004-02, see SE Section 2.4, "Applicable NRC Regulatory Guides, Review Plans, and Guidance Documents."

The SRM associated with SECY-10-113, "Closure Options for Generic Safety Issue 191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance," dated December 23, 2010,[38] directed the NRC staff to consider a risk-informed approach for resolution to GSI-191. In 2012, the staff developed three options to resolve GSI-191. These options were documented and proposed to the Commission in SECY-12-0093.[39] The options are summarized as follows:

- Option 1 allows licensees to demonstrate compliance with 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," through approved models and test methods. This option is expected to be used by low-fiber plants [plants with small amounts of debris] that can demonstrate adequate strainer performance and show that less than 15 grams of fiber per fuel assembly can reach the core.

- Option 2 requires implementation of additional mitigating measures and allows additional time for licensees to resolve issues through further industry testing or use of a risk-informed approach.

- Option 3 involves separating the regulatory treatment of the sump strainer and in vessel effects so that strainer issues can be treated deterministically and in-vessel issues can be risk informed.

These options allowed industry alternative approaches for resolving GSI-191. The Commission issued SRM-SECY-12-0093 on December 14, 2012, [35] approving all three options for closure of GSI-191.

By letter dated June 4, 2012, [40] STPNOC stated that it would pursue Option 2 for the closure of GSI-191 and GL 2004-02, and intended to use a risk-informed methodology.

1.2.4 STPNOC Efforts to Resolve GSI-191 and Respond to GL 2004-02

By letter dated June 19, 2013, as supplemented by letters dated October 3, October 31, November 13, November 21 and December 23, 2013 (two letters); January 9, February 13, February 27, March 17, March 18, May 15 (two letters), May 22, June 25, and July 15, 2014; March 10, and March 25, 2015, STPNOC submitted LARs to allow changes to its licensing basis analysis for certain LOCA events to use a risk-informed approach to resolve NRC concerns associated with the impact of debris blockage on emergency recirculation at STP. The NRC staff conducted several audits and plant trips and submitted several rounds of requests for additional information (RAIs).

By letter dated August 20, 2015, as supplemented by letters dated April 13, May 11, June 9, June 16, July 18, July 21 (two letters), July 28, and September 12, 2016, STPNOC submitted a revised LAR to use a modified risk-informed approach to resolve GSI-191 and close GL 2004-02. The August 20, 2015, submittal described STPNOC's "Risk over Deterministic" (RoverD) methodology. RoverD described STPNOC's revised and simplified methodology that combined both the traditional NRC-accepted deterministic analysis methods and a risk-informed element to form the primary framework. The deterministic analysis used the plant-specific testing results to help determine a threshold at which the risk-informed aspects would be applied. For more details of the STPNOC RoverD methodology refer to SE Section 1.3, "Licensee's Approach - Risk Over Deterministic (RoverD) Methodology."

By letters dated October 20, November 9, and December 7, 2016, and January 19, 2017, STPNOC submitted further revisions to the LAR, updated portions of the previous August 20, 2015, supplement, and responded to NRC staff questions.

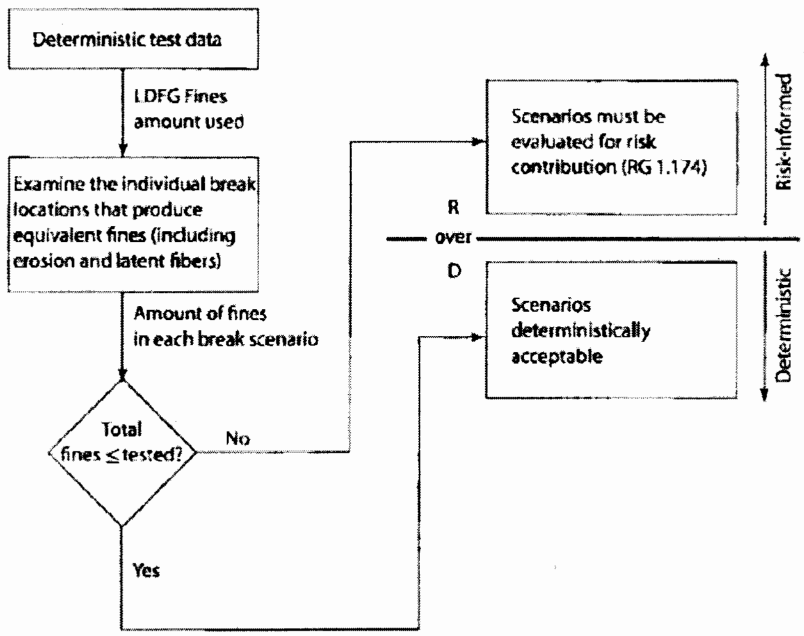
The August 2015 submittal also requested exemptions from certain emergency core cooling and containment cooling requirements contained in 10 CFR 50.46 and 10 CFR 50 Appendix A, General Design Criterion (GDC) 35, Emergency core cooling; and GDC 38, Containment heat removal. The licensee also asked for an exemption from certain containment cleanup requirements in GDC 41, Containment atmosphere cleanup. The licensee stated that the exemptions are necessary to support STPNOC's risk-informed approach to addressing GSI-191 and responding to GL 2004-02 since historically the NRC staff has only accepted deterministic analysis to show compliance with 10 CFR 50.46. The NRC staff granted the exemptions (the transmittal letter can be found in the Agencywide Documents Access and Management System (ADAMS) at Accession No. ML17037C871); the basis for granting the exemptions is found in this SE, Sections 3.0 and 4.0, providing the risk-informed evaluation and conclusions.

1.3 <u>Licensee's Approach - Risk Over Deterministic (RoverD) Methodology</u>

The licensee developed the RoverD methodology to analyze the effects of debris during LOCA events using both the plant-specific testing and risk-informed analysis. The methodology is represented in the flow chart in Figure 1 and uses STP test data to determine a threshold debris value above which ECCS and CSS function may be lost, and also follows the guidance of Regulatory Guide (RG) 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," May 2011,[41] to estimate the risk attributable to debris.

The RoverD methodology evaluated a deterministic category of failures in which the amount of debris [referred to as LDFG (low density fiberglass) fines in Figure 1, or fiber fines] known to cause sump strainer clogging following a LOCA were equal to or less than the amount of debris used in the plant-specific testing for STP. The STPNOC testing used NRC-approved methods to establish an ECCS strainer debris load above which loss of function may occur. This is represented by 'Deterministic test data' in Figure 1. See discussion in SE Section 3.4.2.8.3, "Head Loss and Vortexing," for details on the STPNOC deterministic testing.

Figure 1. STPNOC's RoverD Flow Chart



The RoverD method then used a platform called CASA Grande to examine various break sizes, orientations, and locations to identify the amount of fiber fines generated and transported for each scenario (see SE Section 3.4.2.6.2, "Debris Generation and Zone of Influence," for more details). The results were compared to the tested amount to determine if the scenario is a success under the deterministic testing threshold (ECCS and CSS function is assured) or it exceeds the tested amount and must be categorized as a risk-informed scenario (ECCS and CSS function are assumed to fail).

For the risk-informed scenarios, RoverD quantified the core damage frequency (CDF), change in CDF (ΔCDF), large early release frequency (LERF), and change in LERF (ΔLERF). See SE Section 3.4.2.10, "Systematic Risk Assessment," for the NRC staff's evaluation on risk. These risk values were compared to the acceptance guidelines in RG 1.174. The RG 1.174 guidance was developed in consideration of the Commission's Safety Goal Policy Statements on safety goals and the use of probabilistic risk assessment methods in nuclear regulatory activities (51 FR 30028; August 4, 1986,[42] and 60 FR 42622; August 16, 1995,[43] respectively). Thus, RG 1.174 provides an acceptable method for licensees and NRC staff to use in assessing the impact of licensing basis changes when the licensee chooses to use risk information.

The changes to the methodology used by STPNOC as discussed in Section 1.3 of this SE, including the supplements, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 16, 2016 (81 FR 7843).

1.4 Method of NRC Staff Review

The NRC staff reviewed the licensee's application to ensure that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) activities proposed will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public. The purpose of the NRC staff's review was to evaluate the licensee's assessment of the impact of debris on ECCS and CSS function for design-basis LOCAs. The NRC staff evaluated the licensee's application and supplements. The NRC staff also conducted audits of certain information at the plant and vendor sites, and conducted independent analyses, in areas deemed appropriate by the NRC staff.

In areas where the licensee and its contractors used NRC-approved or widely accepted methods in performing analyses related to the proposed requests, the NRC staff reviewed relevant material to ensure that the licensee/contractor used the methods consistent with the limitations and restrictions placed on the methods. Details of the NRC staff review, audits, and confirmatory calculations are provided in SE Section 3.0, "Technical Evaluation: Risk-Informed Change to the Licensing Basis."

The NRC staff's review of STPNOC's pilot risk-informed resolution to GSI-191 was unique and complex since it involved deterministic and risk-informed reviews across several disciplines. The results of the different elements of the engineering analyses were considered by the NRC staff in an integrated manner such that the merits of each element were considered individually as well as evaluated for impacts on the entire risk-informed project. In this way, the NRC staff decision to approve or deny the STPNOC request was not driven solely by the numerical results of the probabilistic risk assessment (PRA). The PRA was one part of an overall assessment of the proposed changes on the safety of the STP units.

As discussed in SE Section 1.2.3, "Generic Letter 2004-02," licensees were requested to evaluate debris on the sump strainers and in the reactor core (in-vessel) for long-term core cooling impacts. To evaluate the impacts on the sump strainers, the licensee utilized plant-specific testing and complex analysis methods to determine debris amounts for breaks. The plant-specific testing conducted was used to develop conservative threshold values for debris above which ECCS and CSS function was not assured. For the limited scenarios where function was not assured, the licensee quantitatively assessed the change in risk.

For the in-core thermal-hydraulic evaluation, the licensee used a calculational platform that the NRC staff had not previously approved for this kind of application. The acceptance criteria provided in WCAP-16793, Revision 2, "Evaluation of Long Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," July 2013,[44] and the guidance in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for

Nuclear Power Plants: LWR [light-water reactor] Edition," (SRP) Section 15.0.2, "Review of Transient and Accident Analysis Methods," December 2005,[45] provided the bases for the NRC staff analysis in SE Attachment 2. The review of the in-core thermal-hydraulic methodology was exclusively for STP-specific simulations and the results of those simulations.

The NRC staff's review documented below integrates several disciplines (e.g., mechanical, structural, thermal-hydraulic, risk) and this SE is organized using the five key principles of risk-informed regulation as listed in SE Section 3.0.

2.0 REGULATORY EVALUATION

2.1 Description of Affected Structures, Systems, and Components

A fundamental function of the ECCS is to recirculate water that has collected in the containment sumps following a break in the reactor coolant system (RCS) piping to ensure long-term removal of decay heat from the reactor fuel. Leaks from the RCS in excess of the plant's normal make-up capability, scenarios known as LOCAs, are part of every nuclear power plant's design basis. Hence, nuclear plants are designed and licensed with the expectation that they are able to remove reactor decay heat following a LOCA to prevent core damage. Long-term cooling following a LOCA is also a basic safety function for nuclear reactors. The recirculation sump located at the lower areas of the reactor containment structure provides a water source to the ECCS in a PWR once the initial water source has been depleted and the systems are switched over to recirculation mode for extended cooling of the core.

For STP, the ECCS consists of the high head safety injection (HHSI) and low head safety injection (LHSI) pumps, safety injection system accumulators, residual heat removal heat exchangers, and the refueling water storage tank along with the associated piping, valves, instrumentation, and other related equipment.

The CSS consists of the containment spray pumps and the refueling water storage tank along with the associated piping, valves, instrumentation, and other related equipment.

The containment heat removal system consists of the reactor containment fan cooler subsystem, which is a part of the reactor containment heating, ventilating, and air conditioning system, the residual heat removal heat exchangers, and the CSS. The ECCS assists the containment heat removal system by transferring heat from the reactor core to the containment sump. The residual heat removal heat exchangers, in conjunction with the ECCS LHSI pumps, are used to transfer heat from the containment sumps to the component cooling water system. The reactor containment fan coolers are also cooled by the component cooling water system following a safety-injection signal. The component cooling water system rejects this heat to the ultimate heat sink via the essential cooling water system.

Three independent sumps (containment emergency sumps) serve as reservoirs to the ECCS and CSS pumps during the recirculation phase post-DBA. Each sump is stainless steel lined, contains a vortex suppressor, and is provided with four stainless steel strainer assemblies. The strainer assemblies for each sump consist of two 5-module assemblies, one 4-module assembly, and one 6-module assembly with each module made up of 11 strainer discs. The strainer screen consists of a perforated plate with nominal 0.095-inch diameter openings. Flow leaving the strainer enters a four inlet plenum box (one inlet for each strainer assembly). The plenum box collects the flow from the strainer assemblies and directs the flow vertically downward directly into the sump pit. An access cover is provided on the plenum box for internal inspections of the sump structures, vortex suppressor, and the strainer assemblies.

The sumps are located at the minus 11-foot 3-inch elevation level of the reactor containment building. The sumps are physically separated from each other. The floor around the emergency sumps slopes away from them and toward normal sumps located in the area. The drains from the upper levels of the reactor containment building do not terminate in the immediate area of the sumps. The sump structures are designed to withstand the safe shutdown earthquake without loss of structural integrity.

Most potential sources of debris are remote from the emergency sumps and are separated by shield walls or other partitions. Expected debris constituents are pieces of insulation and paint particles.

The potential for paint from structures and components becoming a debris source has been reduced by requiring proper surface preparation and by painting large-surface components (such as the containment liner, RCS supports, floors, and structural steel) with coatings that are qualified for DBA conditions.

Stainless steel reflective metal insulation is used exclusively on the reactor vessel, including the reactor head. Blankets of fiberglass type insulation are used on hot piping, valves, and other equipment including the steam generators, pressurizer, and reactor coolant pumps. Cellular glass insulation is used on cold piping for antisweat purposes. Microtherm® is also used around piping in the wall penetrations.

Containment emergency sumps are inspected periodically as delineated in the technical specifications.

The ECCS components are designed such that a minimum of two accumulators delivering to two unaffected loops, and one HHSI and one LHSI pump delivering to an unaffected loop, will assure adequate core cooling in the event of a design-basis LOCA. The redundant onsite standby diesel generators provide adequate emergency power to all necessary electrically operated components if a loss of offsite power occurs simultaneously with a LOCA, even assuming a single failure in the emergency power system such as the failure of one diesel generator to start.

The ECCS injects borated water into the RCS following a LOCA to limit core damage, metal/water reaction, and fission product release, and to provide, in conjunction with the control rods, sufficient negative reactivity to assure safe shutdown of the reactor core. Borated water can be injected from the accumulators and the refueling water storage tank. The ECCS also provides long-term, post-accident cooling of the core by recirculating borated water from the containment sump to the core.

The ECCS consists of three independent trains, each one capable of providing 100 percent of the required flow to the core following a LOCA. Each train consists of one HHSI pump and one LHSI pump. Heat is removed from the system during recirculation by the residual heat removal heat exchangers (low-head pump only). The piping and valves associated with each of the three subsystems are identical. In the event of a LOCA, the ECCS provides adequate shutdown capability.

Following a postulated LOCA, heat is removed from and pressure is reduced in the containment by the reactor containment fan coolers and the CSS. Component cooling water is circulated through the reactor containment fan coolers and borated water from the refueling water storage tank is sprayed into the containment atmosphere. Long-term post-LOCA containment cooling and pressure reduction is provided by switching the CSS suction to the containment sump, and by the reactor containment fan coolers.

The component cooling water system acts as an intermediate fluid barrier between the radioactive systems and the essential cooling water system to reduce the probability of leakage of radioactivity from the plant to the environment.

The component cooling water system is designed to provide a continuous supply of cooling water to remove heat from the reactor during normal shutdown, to cool the letdown flow to the chemical volume control system during power operation, to cool various engineered safety features heat loads after a postulated LOCA, and to remove heat from various other plant components during normal operation.

2.2 Regulatory Requirements

The NRC staff's acceptance criteria for ECCS performance following a LOCA are based on 10 CFR 50.46. Loss-of-coolant accidents are postulated accidents that would result in the loss of reactor coolant from piping breaks in the reactor coolant pressure boundary at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection and ECCS systems are provided to mitigate these accidents. The NRC staff's review covered the acceptance criteria based on 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria considering the effects of debris as specified in GL 2004-02.

Since STP Units 1 and 2 were built following the promulgation of 10 CFR Part 50, Appendix A General Design Criteria in 1971, the STP licensing basis must satisfy these minimum requirements for design of structures, systems, and components. The NRC staff's review considered GDC 35 requirements for the function of the ECCS, which relies on a properly functioning containment sump. Additionally, the CSS relies on the containment sump for a supply of cooling water for containment heat removal and containment atmosphere clean-up, thus, GDCs 38 and 41 were also considered by the NRC staff during its review.

GDC 35, Emergency core cooling, in part, states:

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

GDC 38, Containment heat removal, in part, states:

> A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature

> following any loss-of-coolant accident and maintain them at acceptably low levels.

GDC 41, Containment atmosphere cleanup, in part, states:

> Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

The following GDCs are considered insofar as they relate to the NRC staff evaluation of the containment design evaluations as impacted by Nuclear Safety Advisory Letters discussed in SE Section 3.4.2.8.5, "Net Positive Suction Head."

GDC 16, Containment design, states:

> Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

GDC 50, Containment design basis, states:

> The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

The NRC requirements for TS are in 10 CFR 50.36(b), which specifies that each license authorizing operation of a utilization facility will include TS and that the TS will be derived from the analyses and evaluations included in the safety analysis report, and amendments. The requirements in 10 CFR 50.36, state that TS are to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls.

2.3 Applicable NRC Regulatory Guides, Review Plans, and Guidance Documents

The Nuclear Energy Institute (NEI) developed an evaluation guidance document titled "PWR Containment Sump Evaluation Methodology," dated May 28, 2004.[46] On December 6, 2004, the NRC issued an SE for that document which found that the NEI document provided an acceptable overall guidance methodology, but that portions needed additional justification and modification.[47] Modifications were made and the final guidance was provided as NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," in December of 2004. Together, Volume 1 of NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," December 2004, and Volume 2, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02," dated December 6, 2004,[48] describe a method acceptable to NRC staff, with limitations and conditions, for performing the evaluations requested by GL 2004-02.

In addition to the evaluation guidance of NEI 04-07, the industry developed the following topical reports (TRs) to aid licensees in responding to GL 2004-02.

- TR-WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," March 2008.[49]
- TR-WCAP-16406-P-A, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," Revision 1, March 2008.[50]
- TR-WCAP-16793-NP-A, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," Revision 2, July 2013.[44]

The NRC staff determined that the reports listed above, subject to the conditions and limitations contained in the NRC SEs for those topical reports, describe a method acceptable to NRC staff for performing the evaluations and analyses within the scope stated in those documents.

To more clearly communicate the NRC staff's expectations for the level of technical detail in the licensees' submittals, the NRC staff issued the "Revised Content Guide for Generic Letter 2004-02 Supplemental Responses," dated November 21, 2007,[51] and the "Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized Water Reactors," dated March 28, 2008 (Content Guide).[52] The Content Guide describes the information necessary to be submitted to the NRC for review in each of the following areas (the section of the SE where the topic is evaluated is identified in parentheses):

- Corrective actions taken to address GL 2004-02 (SE Section 1.2.4, "STPNOC Efforts to Resolve GSI-191 and Respond to GL 2004-02")
- Break selection (SE Section 3.4.2.6.1, "Break Selection")
- Debris generation and zone of influence (SE Section 3.4.2.6.2, "Debris Generation and Zone of Influence")
- Debris characteristics (SE Section 3.4.2.6.3, "Debris Characteristics")

- Latent debris (SE Section 3.4.2.6.4, "Latent Debris")
- Debris transport (SE Section 3.4.2.7, "Debris Transport")
- Head loss and vortexing (SE Section 3.4.2.8.3, "Head Loss and Vortexing")
- Net positive suction head (SE Section 3.4.2.8.5, "Net Positive Suction Head")
- Coatings evaluation (SE Section 3.4.2.6.5, "Coatings")
- Debris source term (SE Section 3.4.2.6, "Debris Source Term")
- Screen modification package (SE Section 3.4.2.8.2, "Screen Modification Package")
- Sump structural analysis (SE Section 3.4.2.8.4, "Sump Structural Analysis")
- Upstream effects (SE Section 3.4.2.8.1, "Upstream Effects")
- Downstream effects – components and systems (SE Section 3.4.2.8.7, "Downstream Effect-Components and Systems")
- Downstream effects – fuel and vessel (SE Section 3.4.2.8.8, "Downstream Effects-Fuel and Vessel")
- Chemical effects (SE Section 3.4.2.8.6, "Chemical Effects")
- Licensing basis (SE Section 2.1, "Description of Affected Structures, Systems, and Components")

For the sump structural analysis, the Content Guide requested a summary of the design inputs, codes, and loads used for the analysis; the structural qualification results and design margins for various components of the structural assembly; the evaluations performed for dynamic effects associated with HELBs; and, a summary statement regarding the structural analysis considering reverse flow if a backflushing strategy is credited.

Section 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments," of the SRP, [53] lists acceptable codes and standards for design of containment internal structures. General guidance for evaluating the technical basis for proposed risk-informed changes is provided in SRP Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance." [54]

RG 1.174 [41] describes an acceptable risk-informed approach for assessing the nature and impact of proposed licensing-basis changes by considering engineering issues and applying risk insights. This regulatory guide also provides risk acceptance guidelines for evaluating the results of such changes. RG 1.174 provides the five key principles of risk-informed integrated decision making.

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities," March 2009, [55] describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results such that the PRA can be used in regulatory decision-making for light-water reactors.

RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 4, March 2012,[56] provides guidance for an evaluation of the effects of debris on ECCS strainers and more generally guidance for the evaluation of water sources for long-term recirculation following a LOCA.

Guidance on evaluating PRA technical adequacy is provided in SRP Chapter 19, Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests After Initial Fuel Load," Revision 3.[57] Section 19.2 of the SRP references the same criteria as RG 1.174, and states that a risk-informed application should be evaluated to ensure that the proposed changes meet the five key principles of risk-informed regulation. Sections 3.1 through 3.5 of this SE contain details of the NRC staff's evaluation of the licensee's proposed changes with respect to the five key principles.

The regulatory requirements and applicable regulatory guides for the long-term core cooling evaluation methodology and results assessment are listed in SE Attachment 2.

3.0 TECHNICAL EVALUATION: RISK-INFORMED CHANGE TO THE LICENSING BASIS

The NRC staff performed its integrated review of the proposed risk-informed approach considering the five key principles of risk-informed decision-making set forth in RG 1.174:

1. The proposed change meets the current regulations, unless it explicitly relates to a requested exemption or rule change.

2. The proposed change is consistent with the defense-in-depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes result in an increase in core damage frequency (CDF) or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

5. The impact of the proposed change should be monitored using performance measurement strategies.

3.1 Key Principle 1: The Proposed Change Meets Current Regulations Unless it is Explicitly Related to a Requested Exemption or Rule Change

The proposed change would modify STP's licensing basis analyses to show compliance with 10 CFR 50.46, considering the effects of debris in containment following postulated accidents by using both deterministic and risk-informed methodologies.

The primary guidance documents used to show regulatory compliance with 10 CFR 50.46 considering the effects of debris by using deterministic criteria are NEI 04-07, WCAP-16793, RG 1.82, and the SRP. A part of the modified STP licensing basis analysis utilizes a traditional deterministic method by using plant-specific testing and other deterministic methods to determine the amount of debris generated and transported for postulated breaks in containment. Using its RoverD method, the licensee concluded that most of the possible break scenarios produced debris amounts that were within the deterministic testing threshold. Thus, the deterministic portions of the RoverD methodology meet the first key safety principle of RG 1.174 since they use NRC staff approved methods and meet the current deterministic evaluation criteria expected for compliance with 10 CFR 50.46. The specific deterministic portions of RoverD are LOCA breaks that may generate and transport debris within the limits established by testing, and, for the in-vessel hot-leg break analysis, hot-leg breaks smaller than 16 inches.

The approval of a risk-informed methodology would require exemptions from 10 CFR 50.46(a)(1)(i) and GDCs 35, 38, and 41, because the NRC has interpreted these regulations as requiring a deterministic approach and bounding calculation to show core cooling acceptance criteria in 50.46(b) are met. The request for one or more exemptions to obtain relief from compliance with the requirement to use deterministic methods is contemplated by the first key safety principle of RG 1.174.

The NRC staff's evaluation of 10 CFR 50.12 exemptions requested from the deterministic requirements of 10 CFR 50.46 and GDCs 35, 38, and 41, for STPNOC to use a risk-informed analysis of post-accident debris effects to demonstrate adequate ECCS and CSS performance, is in a separate document found at ADAMS Accession No. ML17037C871, and is supported, in part, by the evaluation in this SE. The exemptions were granted, in part, because the NRC staff concluded that compliance with a deterministic approach for certain debris effects was not necessary to achieve the underlying purpose of those regulations.

Therefore, the first key safety principle of RG 1.174 is met because the licensee's request for license amendments is related to exemptions requested to use the risk-informed methodology. In addition, appropriate exemptions have been granted.

3.2 Key Principle 2: The Proposed Change Is Consistent with the Defense-in-Depth Philosophy

Section C.2.1.1 of RG 1.174 states that the defense-in-depth philosophy consists of a number of factors, and consistency with the defense-in-depth philosophy is maintained if the following occurs:

1. A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.

2. Over-reliance on programmatic activities as compensatory measures associated with the change in the LB [licensing basis] is avoided.

3. System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).

4. Defenses against potential common-cause failures are preserved, and the potential for the introduction of new common-cause failure mechanisms is assessed.

5. Independence of barriers is not degraded.

6. Defenses against human errors are preserved.

7. The intent of the plant's design criteria is maintained.

In its letter dated August 20, 2015,[20] the licensee provided a discussion of how its risk-informed assessment was consistent with the philosophy of defense-in-depth by addressing each of the seven factors above.

3.2.1 Factor 1: A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation

The licensee stated that the proposed change does not involve equipment or design changes beyond the modifications that have been made in response to the concerns raised in GSI-191, nor does it involve changes to the Emergency Operating Procedures beyond the changes in place to address the concerns raised in GSI-191 (see SE Section 3.4.2.8.2, "Screen Modification Package"). The licensee stated that the proposed change does not significantly affect the containment integrity or the capability of the independent and safety-related reactor containment fan coolers to remove post-LOCA decay heat from containment. The NRC staff reviewed the licensee's information and concludes that factor 1 is met because a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation since the balance that existed before the change is preserved.

3.2.2 Factor 2: Over-reliance on programmatic activities as compensatory measures associated with the change in the licensing basis is avoided

The licensee identified the inservice inspection program, plant personnel training, RCS leak detection program, and containment cleanliness inspection activities as examples of programmatic activities associated with the proposed change. The licensee stated that the proposed change does not rely heavily on programmatic activities as compensatory measures. The licensee also stated that it did not propose any new programmatic activities that could be heavily relied upon. The NRC staff reviewed the licensee's information and concludes that factor 2 is met because there is no over-reliance on programmatic activities as compensatory measures for the risk-informed approach.

3.2.3 Factor 3: System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties

The licensee noted that STP has three independent trains of ECCS equipment for the prevention of core damage, and that the proposed change does not require any design change to these systems. The NRC staff reviewed the licensee's information and concludes that factor 3 is met because system redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties.

3.2.4 Factor 4: Defenses against potential common-cause failures are preserved, and the potential for the introduction of new common-cause failure mechanisms is assessed

The licensee stated that the proposed change does not alter any defenses against common-cause failures. The NRC staff noted that debris has the potential to affect all containment sumps from the same event. The licensee stated that it assessed potential common-cause failure mechanisms, and has in place defenses against potential common-cause strainer clogging, including procedural actions to change flow rates, conserve refueling water storage tank inventory, use alternate injection sources, and to stop or start pumps as necessary. The licensee also noted that the containment fan coolers, which would be unaffected by debris, provide the containment heat removal function in the event the CSS is not available. The NRC staff reviewed the licensee's information and concludes that factor 4 is met because defenses against potential common-cause failures are preserved, and the potential for the introduction of new common-cause failure mechanisms was assessed.

3.2.5 Factor 5: Independence of barriers is not degraded

The licensee considered the three barriers to a radioactive release: the fuel cladding, the RCS piping and components, and the reactor containment building. The licensee stated that it is not changing the design and analysis requirements for the fuel and concluded that the fuel cladding barrier is not changed. The licensee also noted that the very small increase in core damage frequency (CDF) attributable to debris supports a conclusion that this barrier is not significantly degraded. It is described later in this SE that the only events impacted by debris are certain LOCA events as discussed in SE Section 3.4.2.10, "Systematic Risk Assessment"; therefore, in these postulated scenarios, the RCS barrier is breached as a function of the analysis. The licensee stated that the containment function is not directly impacted, but that the containment spray supporting system could be impacted. However, STP has containment fan coolers that can remove containment heat and reduce pressure. Also, the very small increase in LERF attributable to debris, also discussed in SE Section 3.4.2.10, is given as evidence that the containment barrier is not significantly degraded. The licensee thus concluded that debris does not significantly impact the independence of the barriers. The NRC staff reviewed the licensee's information and concludes that factor 5 is met because independence of barriers is not degraded.

3.2.6 Factor 6: Defenses against human errors are preserved

The licensee stated that the proposed change does not involve design changes to the current equipment or changes to operating procedures. This is because operator actions and design changes to address the GSI-191 issues were made before the risk-informed approach was contemplated. The NRC staff notes that the STP risk assessment compared the risk of the as-built plant to a plant with a clean containment. Operator actions are required because debris sources do exist in the containment and defenses against erroneous human actions were addressed under this factor of defense-in-depth. Although the presence of debris has resulted in some new operator actions (e.g., securing one CSS train if all three trains are running), the NRC concludes these additional actions are not excessive in terms of complexity or additional burden on the operators which are contributors to human error. The NRC staff reviewed the licensee's information and concludes that factor 6 is met because defenses against human errors are preserved.

3.2.7 Factor 7: The intent of the plant's design criteria is maintained

The licensee stated that the LAR does not involve changes to the design or design requirements of the current plant equipment associated with GSI-191. A discussion of the plant modifications to address GL 2004-02 prior to 2012 at STP is provided in SE Section 3.4.2.8.2, "Screen Modification Package." The amendment requested allows for changes in licensing basis analysis methods to account for the effects of debris and an associated TS change.

In addition, the licensee requested exemptions from the requirements of 10 CFR 50.46, and 10 CFR Part 50, Appendix A, GDCs 35, 38, and 41 to the extent that those regulations would not allow a risk-informed approach to evaluate post-accident debris effects. The licensee stated that the criteria of 10 CFR 50.12(a)(2)(ii) applied, because compliance in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.

The NRC staff reviewed the exemptions requested by STPNOC and provided its evaluation in a document under ADAMS Accession No. ML17037C871 supported, in part, by this SE. The NRC staff concluded that the special circumstances described in 10 CFR 50.12(a)(2)(ii) exist for each of the exemptions proposed by STPNOC because use of a deterministic approach to evaluate post-accident debris effects is not necessary to achieve the underlying purpose of each rule, which is adequate ECCS and CSS performance.

The NRC staff reviewed the licensee's information and concludes that factor 7 is met because the intent of the plant's design criteria is maintained by means of an alternate non-deterministic evaluation method.

3.2.8 Additional Defense-in-Depth Considerations

To augment the discussion about the seven defense-in-depth elements in RG 1.174, the licensee provided additional information about systems, structures and components (SSCs); plant programs; and design features that support the defense-in-depth philosophy by minimizing the likelihood and consequences of a LOCA and by ensuring adequate containment performance even during events where debris is generated. These are described in the sections below.

3.2.8.1 Reactor Coolant System Weld Mitigation

The licensee noted that with the exception of the welds that connect the reactor vessel nozzles to hot- and cold-leg piping, all large-bore RCS welds susceptible to primary water stress-corrosion cracking (PWSCC) have been either replaced with Alloy 690 material (e.g., steam generator nozzles) or have been overlaid with Alloy 52/52M/152 material (e.g., pressurizer piping safe ends). Both Alloy 690 and Alloy 52/52M/152 materials are less susceptible to PWSCC than the previously used Alloy 600 and Alloy 82/182 materials.

The licensee stated that the reactor vessel nozzle welds are less of a concern in the GSI-191 analysis than other break locations because the reactor vessel is covered with reflective metal insulation, and the primary shield wall would protect the majority of fiberglass insulation in the steam generator compartments from the effects of a LOCA jet. The reactor vessel nozzle welds met the deterministic testing debris limits in the July 2008 test.* The licensee analyzed separately the impact of debris generation of 45 welds determined to exceed the deterministic debris limit using the risk-informed analysis. However, the reactor vessel nozzle welds are assumed to fail in order to simplify the thermal-hydraulic evaluation for in-vessel effects (see Attachment 2, In-Vessel Thermal-Hydraulic Analysis, of this SE for the NRC staff's evaluation of

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* The July 2008 testing was used by STPNOC to provide the debris threshold between the deterministic and risk evaluations for the RoverD methodology; see discussions in SE Section 3.4.2.6, "Debris Source Term," and SE Section 3.4.2.8, "Impact of Debris."

in-vessel effects). These eight welds were combined with the 45 welds that did not meet the deterministic criteria and were included in the risk-informed analysis.

In its letter dated May 11, 2016,[22] the licensee noted that the reactor vessel nozzle welds are the full penetration butt welds that join the hot- and cold-leg piping to the reactor vessel nozzles. The reactor vessel nozzle welds are fabricated with Alloy 82/182 filler material and have not been mitigated. The licensee intends to mitigate them by a process known as the "Mechanical Stress Improvement Process" in spring 2017 and fall 2019 at STP Units 1 and 2, respectively. The Mechanical Stress Improvement Process compresses the pipe adjacent to the weld and creates compressive stresses through approximately 50 percent of the pipe wall thickness. The licensee stated that this process will eliminate the tensile stresses at the inside surface of the PWSCC-susceptible weld and thereby remove one (i.e., tensile stresses) of the three conditions required to be present for PWSCC to initiate.

The licensee replaced the reactor vessel heads in STP Units 1 and 2 in 2009 and 2010, respectively. The licensee stated that the control rod drive mechanism nozzles in the replacement reactor vessel closure head and associated welds are made of Alloy 690/52/152 material and are not considered susceptible to PWSCC. The licensee stated that the breaks associated with the relatively small size and location of the control rod drive mechanism nozzles are not considered in the break selection because they are bounded by larger breaks that produce much more debris.

The licensee noted that primary loop RCS piping and branch connection pipes are constructed of stainless steel (e.g., SA-351 Grade CF8A for RCS loop piping and SA-312 for branch piping) and are not susceptible to PWSCC.

The licensee reported that the RCS piping welds considered for GSI-191, which have been mitigated with material not susceptible to PWSCC are: steam generator nozzles, pressurizer nozzles (overlays), and reactor vessel closure head penetration nozzles. The RCS piping welds considered for GSI-191, which are susceptible to PWSCC and which have not been mitigated, are reactor pressure vessel nozzle welds.

3.2.8.2 Reactor Coolant System Leakage Detection Program

The licensee stated that the RCS leakage detection program is capable of early identification of RCS leakage in the piping inside the containment in accordance with RG 1.45, Revision 1, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," May 2008.[58] The licensee also stated that the early detection provides time for appropriate operator action to identify and address RCS leakage. The licensee noted that the effectiveness of the RCS leakage detection systems is not reduced by the proposed licensing basis change to the risk-informed approach for GSI-191.

In accordance with industry guidance NEI 03-08, Revision 2, "Guidelines for the Management of Materials Issues," January 2010,[59] STPNOC has implemented a boric acid control program, which requires visual examinations to look for boric acid residue. This program provides early detection of cracking in subject piping to minimize potential of pipe break that may cause debris generation.

3.2.8.3 Plant Modifications

The NRC staff also noted that several plant modifications to improve safety were made prior to this LAR including a significant increase in strainer surface area and removal of problematic materials from containment. A detailed description of these actions was provided to the NRC in the licensee's RoverD submittal dated August 20, 2015.[20] A summary of the changes made to the strainer is included in SE Section 3.4.2.8.2, "Screen Modification Package."

3.2.8.4 NRC Staff Review of Additional Defense-in-Depth

The NRC staff reviewed the licensee's additional defense-in-depth actions and programs discussed above and concludes that the RCS weld mitigation, RCS leakage detection program, and plant modifications completed prior to 2012 to address GL 2004-02 provide additional defense-in-depth measures beyond the seven factors defined in RG 1.174.

3.2.9 NRC Staff Conclusion Regarding Defense-in-Depth

The NRC staff reviewed the licensee's actions and programs relied upon to maintain adequate defense-in-depth in accordance with the seven factors of RG 1.174. The NRC staff concludes that because the licensee has adequately addressed the seven factors of the defense-in-depth philosophy and the licensee has taken additional actions that provide defense-in-depth, this element is satisfied.

3.3 Key Principle 3: The Proposed Change Maintains Sufficient Safety Margins

RG 1.174 states that safety margins are maintained when codes and standards or their alternatives approved for use by the NRC are met and when the safety analysis acceptance criteria in the licensing basis (e.g., UFSAR, supporting analyses) are met or proposed revisions provide sufficient margin to account for analysis and data uncertainty.

The licensee identified margins and conservatisms in the design, analysis, construction, and operation of the plant to show that the proposed methodology change (i.e., risk-informed approach) will maintain sufficient safety margins. Specifically, the licensee identified applicable codes and standards, or the pertinent NRC-approved alternatives; explained how STP met the safety analysis acceptance criteria in its licensing basis; and discussed how proposed revisions provide sufficient safety margin to account for analysis and data uncertainty. The following examples were identified by the licensee and are addressed in SE Sections 3.3.1 through 3.3.6 below:

- Fabrication, Design, and Construction
- American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Class 1 Inservice Inspection and Testing Program
- Debris Threshold Methodology
- Debris Generation and Transport Methodology
- Strainer Testing Methodology

- Plant-Specific Configuration Conservatisms

3.3.1 Fabrication, Design, and Construction

As part of GSI-191 resolution, the licensee postulated breaks in pipes in the containment to demonstrate the performance of the containment strainers and sump capacity. However, the NRC staff noted that the licensee described ASME Code requirements for design, fabrication, construction, and examination of containment piping, and addressed associated safety factors.

The NRC staff notes that as part of the licensing basis in UFSAR Sections 3.2 and 3.6, the licensee demonstrated the structural integrity of all Class 1 pipes in the containment by structural analyses based on the requirements of ASME Code, Section III or construction code at the time of plant construction. The construction, design, and operation of Class 1 piping must satisfy regulations in 10 CFR 50.55a(a), (b), (c), (f), and (g); GDC 4; GDC 14; and GDC 31. GDC 4 provides criteria for environmental and dynamic design, GDC 14 provides criteria for RCS pressure boundary design, fabrication, erection and testing, and GDC 31 specifically requires the RCS piping to maintain safety margins. The NRC staff further notes that the licensee did not take exceptions or request exemptions from the NRC regulations with regard to the fabrication, design, construction, and examination of piping systems as part of its effort to address GSI-191 issues.

The licensee stated that RCS piping at STP is fabricated with stainless steel and welds are made of stainless steel, nickel-based Alloy 82/182 weld metal, or mitigated with nickel-based Alloy 52/152 steel. Based on laboratory tests and operating experience, the stainless steel, whether used in the pipe base metal or in welds, is less susceptible to PWSCC than Alloy 600/82/182 metal. Also, Alloy 52/152 metal is less susceptible to PWSCC than Alloy 82/182 welds. The NRC requires augmented inspection for Alloy 600 components and Alloy 82/182 welds as specified in 10 CFR 50.55a(g)(6)(ii)(E), and 10 CFR 50.55a(g)(6)(ii)(F).

The licensee did not request alternatives per 10 CFR 50.55a(z) to the augmented inspections in the NRC regulations. As stated above, the licensee plans to apply the Mechanical Stress Improvement Process to mitigate potential PWSCC at the reactor vessel nozzle welds that are fabricated with Alloy 82/182 metal in the next few years. The NRC staff notes that many nuclear plants have implemented the Mechanical Stress Improvement Process to mitigate stress-corrosion cracking of susceptible weld material with acceptable results.

The NRC staff finds that the ASME Code Class 1 piping considered in the debris generation analysis is fabricated or mitigated with material that is resistant to cracking such that catastrophic pipe breaks would not likely occur. If cracking does occur, the RCS leakage detection systems will be able to detect leakage and the operator will take corrective actions in accordance with the requirements of STP technical specifications. The NRC staff finds that the subject piping maintains defense-in-depth and safety margin because it satisfies the regulations of 10 CFR 50.55a, GDC 4, GDC 14, and GDC 31. Therefore, the NRC staff concludes that the piping considered in the debris generation analysis will not likely result in a catastrophic double-ended guillotine break that would significantly affect the containment sump performance.

3.3.2 ASME Code Class 1 Inservice Inspection and Testing Program

The licensee stated that the integrity of the ASME Code Class 1 welds, piping, and components are maintained at a high level of reliability through the ASME Code, Section XI, inservice inspection program. The ASME Code, Section XI inservice inspection program ensures that the following requirements of Technical Specification (TSs) 4.0.5 and 4.4.10 have been satisfied:

(a) completion of the inservice inspection of piping and component welds in accordance with the schedule requirements of the 2004 Edition, no Addenda, of the ASME Code, Section XI;

(b) completion of inservice inspection of piping and equipment, and component supports (excluding snubber assemblies), and of containment metal liner in accordance with the schedule requirements of the ASME Code, Section XI; and,

(c) completion of the examinations of the reactor coolant pump flywheels in accordance with the requirements of RG 1.14.[60]

Section 10 CFR 50.55a requires the licensee to periodically inspect the RCS pressure boundary, including piping in accordance with the ASME Code, Section XI.

Section 10 CFR 50.55a(g)(6)(ii)(E) requires components fabricated with Alloy 600/82/182 metal to be inspected in accordance with ASME Code Case N-722-1, "Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated With Alloy 600/82/182 Materials Section XI, Division 1," as conditioned. Code Case N-722-1 requires visual examinations of Alloy 600/82/182 materials periodically.

Section 10 CFR 50.55a(g)(6)(ii)(F) incorporates by reference, with conditions, ASME Code Case N-770-1, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities, Section XI, Division 1." ASME Code Case N-770-1 requires licensees to perform augmented inspections beyond those required by the ASME Code, Section XI, for piping with Alloy 82/182 dissimilar metal butt welds.

The licensee's NRC-approved risk-informed inservice inspection program requires inspection of safety-significant welds in the RCS piping.

The ASME Code, Section XI, IWB-5000 requires the licensee to perform a VT-2 visual examination of potential leakage after the system leakage test during every refueling outage.

The ASME Code and associated Code Cases for Class 1 components have been developed to provide assurance that components that are fabricated and inspected to meet the Code have a low probability of failure. The NRC staff verified that the licensee's implementation of the Codes is appropriate to maintain piping integrity.

The NRC staff finds acceptable that (1) the licensee will follow the applicable requirements of the ASME Code, Section XI and 10 CFR 50.55a(g)(6)(ii) to perform required inservice inspections of the Class 1 pipes, welds, pipe supports, and containment metal liners to monitor their structural integrity, and (2) the licensee will examine the reactor coolant pump flywheels in accordance with the requirements of RG 1.14. The NRC staff concludes that these examinations will ensure low probability of pipe and component failures, thereby, minimize the potential for LOCAs and debris generation.

3.3.3 Debris Threshold Methodology

The licensee stated that the methodology it used to establish the debris limits for the strainer contains conservatism. Conservatism is included in the test methods in the staff review guidance regarding GL 2004-02 closure,[52] which was developed to provide reasonable assurance that test results are bounding for the plant-specific conditions represented in the test. This conservatism is included in NRC-approved deterministic guidance and methodologies to ensure that a licensee accounts for uncertainties. The margins in the evaluation help to ensure that the scenarios that are calculated to be successful would result in adequate long-term core cooling. The licensee used guidance in RG 1.82, NEI 04-07 and the associated NRC staff SE,[48] and NRC staff's review guidance regarding GL 2004-02 closure.[52] NRC guidance was issued in the areas of strainer head loss and vortexing, coatings, and chemical effects as noted in Section 2.4, "Applicable NRC Regulatory Guides, Review Plans, and Guidance Documents," above. These guidance documents contain several conservatisms. As discussed in the specific sections on these topics, the licensee implemented the guidance correctly or used acceptable alternative methods.

3.3.4 Debris Generation and Transport Methodology

The NRC guidance in RG 1.82, NEI 04-07 and the associated NRC staff SE,[48] and NRC staff's review guidance regarding GL 2004-02 closure[52] for debris generation and transport, was developed to ensure that the debris load predicted to reach the strainers is maximized considering plant-specific conditions. Conservatisms in the licensee's evaluation for debris generation and transport include the fact that the sizes of the critical breaks were established using the most conservative break jet orientation. That is, the analysis involved rotation of the break jet around the pipe to identify which direction would result in the largest amount of debris. This effectively minimized the break size considered to result in a failure of the system. Smaller breaks are more likely to occur, therefore, the risk associated with each break was maximized.

3.3.5 Strainer Testing Methodology

The NRC strainer testing guidance in RG 1.82, NEI 04-07 and the associated NRC staff SE,[48] and NRC staff's review guidance regarding GL 2004-02 closure[52] was developed to provide reasonable assurance that head losses predicted from testing represent the most limiting conditions for the plant conditions being tested. The guidance also directs that the application of the test results be performed conservatively. The licensee's test program used the maximum particulate and chemical debris loads that could be generated and transported by any break when establishing the fiber limit used to determine the risk associated with debris.

The licensee performed testing to determine the amount of fiber that could penetrate the strainer. The testing was performed using different test conditions to ensure that the potential for penetration was estimated conservatively. The licensee developed a model based on the testing and performed sensitivity studies that demonstrated that the amount of fiber that may reach the core is low, even using very conservative assumptions.

3.3.6 Plant-Specific Conservatisms

The licensee incorporated other conservatisms into its analysis based on plant-specific attributes. For example, STP has three independent and redundant trains of ECCS. When calculating the amount of fiber that may penetrate the strainers and transport to the reactor core, the licensee studied the effects of the number of strainers and pump configurations to determine the maximum amount of debris that could reach the core. When performing head loss calculations, the licensee assumed that only two strainers would be in service. The licensee also appropriately accounted for scenarios when only one strainer would be in service. This maximizes the debris load on the strainer resulting in a greater number of breaks and minimizing the break sizes that could result in a failure of the cooling function.

3.3.7 NRC Staff Conclusion Regarding Safety Margins

The NRC staff concludes that, as discussed above, the licensee's evaluation includes several methodology and other conservatisms that preserve safety margins and help to assure that the analysis results in a conservative prediction of the plant response to debris-generation scenarios.

3.4 Key Principle 4: When Proposed Changes Result in an Increase in CDF or Risk, the Increases Should Be Small and Consistent with the Intent of the Commission's Safety Goal Policy Statement

This section discusses the licensee's base PRA model, including the calculated total risk values (CDF and LERF) for each unit and the licensee's risk-informed assessment of debris. A review of this information was necessary in order to determine whether the risk attributable to debris is small and consistent with the Commission's Safety Goal Policy Statement.

3.4.1 NRC Review of the Base PRA Model

As stated in RG 1.174,[41] the licensee's PRA should be commensurate with the safety significance of the proposed change and the role the PRA plays in justifying the change. The objective of the NRC staff's review of the base PRA model was to determine whether the STPNOC PRA used in evaluating the risk attributable to debris was of sufficient scope, level of detail, and technical adequacy for this application.

In its letter dated November 13, 2013,[4] the licensee provided information regarding the scope, level of detail, and technical adequacy of its PRA model. In its letter dated October 20, 2016,[30] the licensee described the simplified, bounding approach it used to estimate the risk attributable to debris. The NRC staff's review of this information focused on the ability of the licensee's PRA model to analyze the risks stemming from the presence of debris in containment and did not involve an in-depth review of the licensee's PRA. See SE Section 3.4.1.3, "Technical Adequacy of the Base PRA," for further discussion.

The discussion below considers the scope, level of detail, and technical adequacy of the STPNOC base PRA model. Because the licensee chose to use a simplified, bounding risk assessment, the role played by the licensee's base PRA model was greatly reduced compared to typical risk-informed LARs. The primary roles of the STPNOC base PRA in the licensee's risk analysis were (1) to provide an estimate of the relative frequency of accidents with only one containment sump in operation, compared to scenarios where two or all three sumps operate; and (2) to provide a basis for estimating the LERF split fraction; that is, the ratio of LERF attributable to debris to the CDF attributable to debris. The STPNOC base PRA was also used to identify and screen scenarios that involve operation of the ECCS and containment sprays in recirculation mode; that is, to support the initial screening (plant-wide) as described later in this SE.

3.4.1.1 Scope of the Base PRA (Modes/Hazards)

In its letter dated November 13, 2013,[4] the licensee stated that its PRA is an at-power, Level 1 and Level 2 model that includes external and internal hazards such as internal floods, seismic events, internal fires, high winds, and external flooding. In its letter dated September 12, 2016,[29] the licensee provided mean values (per unit) for the total CDF and LERF (7.9E-6 and 4.7E-7 per year, respectively). The total baseline CDF and LERF are inputs into the RG 1.174 risk acceptance guidelines.

The NRC staff reviewed the licensee's information regarding the scope of its base PRA and concludes that the at-power risk bounds the shutdown risk of debris. Moreover, the licensee's simplified risk assessment, described later in this SE, makes limited use of the base PRA model. The NRC staff concludes that the scope of the licensee's base PRA is sufficient for this application and is consistent with the guidance in RG 1.174, which states that the necessary sophistication of the evaluation, including the scope of the risk assessment, depends on the contribution the risk assessment makes to the integrated decision-making.

3.4.1.2 Level of Detail of the Base PRA

The licensee's simplified, bounding risk assessment, as described in its letter dated October 20, 2016,[30] uses the base PRA in a limited manner, as stated above. The NRC staff reviewed the licensee information and concludes that the level of detail of the licensee's base PRA is sufficient because of its limited role in the simplified analysis used.

3.4.1.3 Technical Adequacy of the Base PRA

Revision 2 of RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009,[55] provides regulatory guidance for assessing the technical adequacy of a PRA. Per RG 1.200, Revision 2, evaluation of the technical adequacy of a PRA relies heavily on the industry peer review of the PRA model against the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard. RG 1.200, Revision 2, states, in part, that:

> When used in support of an application, this regulatory guide will obviate the need for an in-depth review of the base PRA by NRC reviewers, allowing them to focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application.

Therefore, the NRC staff relied on the peer review findings in its determination of the technical adequacy of the base PRA model.

In its letter dated November 13, 2013,[4] the licensee stated that the internal events portion of its PRA was peer reviewed and that it met the guidance in RG 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," January 2007.[61] The licensee provided a list of the peer review findings and noted that all but three findings were determined to be dispositioned. The three findings that were not fully dispositioned were determined by the licensee not to be relevant to the risk-informed resolution of GSI-191. In its letter dated June 25, 2014,[16] the licensee also stated that the peer review conclusions for the internal events portion of the PRA model are applicable to RG 1.200, Revision 2, for this application. The licensee stated that the seismic and fire portions of the PRA have minimal impact to the analysis of risk attributable to debris and thus the technical adequacy of these portions of the model were not addressed.

In its letter dated July 15, 2014, [17] the licensee provided information regarding PRA model changes made after its PRA was peer reviewed. The licensee stated that an independent focused-scope peer review was performed following an upgrade to the human reliability analysis and that all the findings had been resolved. The licensee also stated that a focused-scope independent review was done of the electric power recovery analysis and stated that all the findings had been resolved.

The NRC staff reviewed the licensee's resolution of the findings from the peer review of the internal events PRA and from the focused-scope peer reviews of the model upgrades described above. Based on this review, the NRC staff concludes that the technical adequacy of the licensee's base PRA is acceptable for this application for two reasons. First, aspects of the PRA relied-upon to analyze the risk attributable to debris were peer-reviewed and found to meet Capability Category II of the ASME PRA Standard, as endorsed by RG 1.200. Second, for aspects of the PRA where findings were identified (e.g., uncertainty analysis for station blackout sequences), the licensee provided a technical justification for why the calculation of the risk attributable to debris would not be affected.

3.4.1.4 NRC Staff Conclusion Regarding the Base PRA Model

As discussed above, the NRC staff concludes that the base PRA model used in support of the proposed LAR has the appropriate scope, level of detail, and technical adequacy to analyze the risks attributable to debris, consistent with the guidance in RG 1.174.

3.4.2 Risk-Informed Approach for Addressing the Effects of Debris on Long-Term Core Cooling

Information from the licensee's PRA was combined with traditional engineering analyses to estimate the risk attributable to debris. This integrated analysis is referred to as the "systematic risk assessment."

3.4.2.1 Scope of the Systematic Risk Assessment

The licensee provided information regarding the final scope of its systematic risk assessment which used a successive screening process. This method is common practice in quantitative risk assessments to determine the scope of a systematic risk assessment. The licensee's initial plant-wide screening considered all modes and all hazards to determine which scenarios might contribute to the debris risk. A subsequent location-specific screening was then completed to identify accident sequences that could be adversely impacted by debris. This two-step screening process is described in more detail below.

3.4.2.1.1 Initial Plant-Wide Screening

In its letters dated November 13, 2013, [4] and July 21, 2016, [26] the licensee described a screening approach that was used to identify hazards and scenarios to be considered in the assessment of risk attributable to debris. Relevant scenarios retained for detailed assessment met all of the following criteria:

- The scenario response model for the initiator includes taking credit for recirculation to provide core cooling.
- The scenario involves the potential to liberate a significant amount of insulation inside primary containment.
- The scenario includes a mechanism that transports the liberated insulation debris to the sump(s).
- In the absence of GSI-191 phenomena, the scenario would have been evaluated as successfully terminated.

The licensee screened out any hazard or scenario not satisfying all four criteria. The licensee applied the criteria to internal events such as reactor coolant pump seal LOCA, non-isolable small-break loss-of-coolant accident (SBLOCA), open safety relief valve, pipe-break LOCAs, steamline break inside and outside containment, and other transient initiators including support system failure initiators. The licensee concluded that the only initiating events fulfilling the four criteria were medium and large-break loss-of-coolant accidents (LBLOCAs). The licensee also considered internal plant fires and internal plant floods, as well as external hazards (e.g., high winds, seismic events), and concluded that they did not merit further consideration because they failed to meet criteria 2 and 3. The licensee concluded that only medium and LBLOCAs warranted further consideration with respect to GL 2004-02. The licensee included a summary of each scenario that was screened and the basis for each screening determination.

In its letter dated May 22, 2014,[15] the licensee provided a simplified assessment of seismic LOCA risk. Using site-specific seismic hazard curves that were developed to address recommendations made in response to the Fukushima event and generic LOCA fragility curves developed by the Electric Power Research Institute, the licensee computed that the frequency of LOCAs initiated by seismic events would be about 1E-7 per year for medium LOCAs and about 5E-8 per year for LBLOCAs. In its letter dated October 20, 2016,[30] the licensee stated that only LOCAs in the large range produce sufficient quantities of debris to challenge ECCS and CSS performance (discussed further in SE Section 3.4.2.1.2, "Location-Specific Screening"). The licensee's assessment did not differentiate between direct and indirect seismic LOCA events.[†] The licensee concluded that seismically induced LBLOCAs may be screened from further consideration. The NRC staff reviewed the licensee's information and concludes that screening seismically induced LBLOCAs from further consideration is acceptable because seismically induced LBLOCA events are a small fraction of the other LOCA events that are retained in the analysis.

In its letter dated July 28, 2016,[28] the licensee provided a different analysis of the contribution of LOCAs caused both directly and indirectly by seismic events. The licensee used the methods referenced in NUREG-1903, "Seismic Considerations for the Transition Break Size," February 2008,[62] for these analyses. Regarding direct seismic LOCAs, the licensee quoted the

[†] A "direct" seismically induced LOCA involves rupture of a piping or non-piping component caused by the seismic event itself. An "indirect" seismically induced LOCA is caused by, for example, failure of piping or component supports that leads to the consequential failure of the piping or non-piping component.

conclusion from NUREG-1903 that "...the probability of a direct DEGB [double-ended guillotine break] in RCS piping is very low." The licensee performed additional calculations consistent with NUREG-1903 for indirect seismic LOCAs and concluded that the frequency would be on the order of 10^{-10} to 10^{-9} per year, not including station blackout scenarios. The licensee did not include instances of seismically induced LOCAs with simultaneous station blackout because (1) those cases are judged to directly lead to core damage (independent of debris effects), and (2) power would not be available to the ECCS or containment spray pumps. The licensee concluded that indirect seismically induced LOCA events may be screened from the systematic risk assessment.

In its letter dated July 21, 2016,[26] the licensee analyzed the possibility of LOCAs caused by water-hammer events and identified plant-specific design and operational measures relied upon to prevent them. The licensee also provided plant-specific operational experience (e.g., corrective action program data) to demonstrate that these measures have been effective in preventing water-hammer events.

In its November 13, 2013, and July 21, 2016, letters, the licensee evaluated which plant operating conditions should be considered in the systematic risk assessment. The licensee assumed that a LOCA that occurs during full power operation is equivalent or bounding compared to the other operating modes. The reasons given were that the RCS pressure and temperature—key inputs affecting the zone of influence (ZOI) of a LOCA jet—would either be approximately the same or significantly lower for non-power modes. Also, the flow rate required to cool the core—a key input affecting core blockage—would be significantly reduced for low power or shutdown modes.

The licensee's initial screening process concluded that only medium and LBLOCA events warranted further analysis with respect to the risk attributable to debris and that full power operation bounded other plant operating states. The NRC staff reviewed the licensee's approach for determining which hazards, initiating events, and operating modes should be considered in the systematic risk assessment, and concludes that only medium and LBLOCA events occurring at full power need to be considered as a result of the initial screening because the key inputs (i.e., temperature, pressure, flow) affecting debris generation and transport would be less significant for lower or non-power modes.

3.4.2.1.2 Location-Specific Screening

The licensee's initial screening process included all medium or LBLOCA scenarios that could generate and transport any amount of debris to the sump during recirculation and would not otherwise lead to core damage. In order to further refine the scope of the risk-assessment, the licensee performed a number of analyses to characterize scenarios of interest and to determine how much debris is generated and transported, the effect of debris on sump strainer performance or the reactor core, etc. At a high level, these analyses included the following steps:

1. List all possible break locations (e.g., pipe welds, valve bodies, manways, etc.) that could support a medium or LBLOCA.

2. Justify retaining only pipe-break LOCAs occurring at weld locations as important to the risk calculation.

3. For each location, determine the amount and characteristics of debris that could be generated (i.e., type of debris, size distribution).
4. Determine the amount of generated debris that would transport to the sump.
5. Conduct a test to empirically determine a conservative threshold amount of debris, below which failure of ECCS or CSS equipment would not occur.
6. Compare the amount of debris predicted to reach the sump to the threshold value determined by testing.
7. Screen from further consideration (i.e., "negligible" risk contribution) break locations that transported less than the threshold value.
8. Retain break locations that transported more than the threshold value for further analysis and risk quantification.

Certain break locations were scoped into the risk assessment, although the analysis showed the maximum debris value as less than the strainer threshold. This was done to simplify the deterministic in-core thermal-hydraulic analysis of debris for hot-leg LBLOCAs. Additional detail regarding these break scenarios is discussed in SE Section 3.4.2.10, "Systematic Risk Assessment," and SE Attachment 2, "In-Vessel Thermal Hydraulic Analysis."

The licensee's screening process results showed that the risk assessment of debris was limited to LBLOCAs occurring at a subset of the primary system welds.

The licensee initially used guidance from NEI 04-07 to perform a deterministic break selection process. In general, the NEI 04-07 break selection methodology determines a few break locations that result in the limiting debris load or loads that can reach the strainer. The method is described in more detail in SE Sections 3.4.2.6, "Debris Source Term," and 3.4.2.6.1, "Break Selection." The licensee later determined that the maximum debris loads predicted by the break selection method were beyond those that the strainer could accommodate. In particular, the licensee determined that the fibrous debris amount predicted to arrive at the strainer was too great. The licensee concluded that all other debris types could be accommodated in the full amounts if the fibrous debris amount was limited. In order to determine which initiating events generated and transported debris loads that could be accommodated by the strainers, the licensee performed a break location-specific evaluation for all ASME Class 1 piping welds in the RCS. The licensee's methodology had already screened other initiating events. It also assigned the frequency of non-weld break locations (e.g., manways) to the piping welds.

The licensee proposed that any LOCA break that generates and transports more fine fibrous debris to the strainer than can be accommodated by the strainer is assumed to result in core damage. Breaks that result in less than the acceptable amount of fine fibrous debris reaching the strainer are considered to result in successful operation of the ECCS and CSS strainers.

The acceptable fine fiber debris loading was determined by the testing described in SE Section 3.4.2.8.3, "Head Loss and Vortexing." Other debris types were included in bounding amounts in the test. Therefore, the location-specific screening was conducted based on the amount of fine fiber that was calculated to arrive at the strainer.

In order to determine the amount of fiber generated from each potential break location, the licensee developed a computer assisted design (CAD) model of containment. The CAD model included the locations of each potential break location and the fibrous insulation that could be damaged by a LOCA jet. The licensee also calculated the amount of fine fiber that would transport to the strainer considering the amount of fiber generated by each break. Each break location was evaluated by the combination of the debris generation and transport models to determine the largest amount of debris that could arrive at the strainer due to a break at that location. The licensee's implementation of the CAD model for debris generation calculations is discussed in SE Section 3.4.2.6.2, "Debris Generation and Zone of Influence." A description of the transport methodology is provided in SE Section 3.4.2.7, "Debris Transport."

In addition to breaks that may result in debris loading in excess of that which can be accommodated by the strainer, the licensee assumed that reactor hot-leg nozzle breaks which do not generate debris in excess of the strainer limits, will still result in core damage. This assumption was made to simplify the licensee's thermal-hydraulic evaluation of the effects of debris blockage at the core inlet. These weld locations were combined with the specific break locations that result in debris generation in excess of the established limit. The reactor nozzle breaks and the breaks that result in debris generation and transport greater than the limit are the individual locations that are considered by the licensee to result in core damage in the risk evaluation.

The NRC staff concludes that the licensee's location specific screening is acceptable because the process resulted in the identification of all locations that could result in a failure of the ECCS long-term core cooling function due to debris effects.

<u>NRC Staff Conclusion Regarding the Scope of the Systematic Risk Assessment</u>

The NRC staff reviewed the licensee's plant-wide screening information and, for the reasons discussed above concludes that only medium and LBLOCA events need to be considered and that full-power operation bounds other plant operating states because the licensee considered an appropriate range of hazards and plant operating conditions and used an appropriate systematic screening process.

3.4.2.2 Initiating Event Frequencies

In its letter dated October 20, 2016,[30] the licensee discussed how initiating event frequencies were determined. In order to assign frequencies to the LOCAs that screened into the systematic risk assessment, the licensee used values from NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," April 2008, Volumes 1 and 2.[63] The guidance in NUREG-1829 provides exceedance frequencies for discrete break sizes; that is, the annual probability of having a specified break at a given size or larger.

The licensee used linear interpolation to determine frequencies for break sizes not specifically listed in NUREG-1829. The guidance in NUREG-1829 states that interpolation may be used but does not specify any one interpolation scheme. The licensee provided a sensitivity study that included results from linear interpolation and log-log interpolation. This analysis showed little sensitivity of the final result to the choice of interpolation scheme; however, linear interpolation was conservative (i.e., it produced higher LOCA frequencies). The NRC staff finds the licensee's use of linear interpolation of the NUREG-1829 data acceptable because it is conservative relative to other interpolation methods that are commonly used for curves of this sort (e.g., log-log or log-linear, etc.).

The NUREG-1829 guidance contains "25-year" or "current" LOCA frequencies and "40-year" or "end of license period" LOCA frequencies. For most LOCA types, the 40-year values are slightly higher due to anticipated aging effects and the possibility of new degradation mechanisms. In some cases, however, the 40-year values are lower, reflecting an expectation that improved mitigation techniques will lower LOCA frequencies. NUREG-1829 recommends the use of the 25-year values for plants that have been operating between 25 and 40 years. STP Units 1 and 2 were licensed in 1988 and 1989, respectively, and therefore have been operating between 25 and 40 years. The licensee used "25-year" LOCA frequencies from NUREG-1829. The NRC staff concludes that this is acceptable because it is consistent with the assumption in NUREG-1829.

The NRC staff noted that the frequencies in NUREG-1829 apply only to LOCAs caused by long-term material degradation. As discussed in SE Section 3.4.2.1.1, "Initial Plant-Wide Screening," the licensee addressed both seismic and water-hammer events as possible LOCA initiators, and was able to screen those events from further analysis.

The NUREG-1829 guidance provides LOCA frequencies that are based on a formal expert elicitation process. Several aggregation schemes are presented that combine, or aggregate, the inputs of the individual experts into a single set of frequencies that can be used for decision-making. The two primary aggregation schemes are the geometric mean and simple average or arithmetic mean. Because alternate aggregation methods can lead to significantly different results, NUREG-1829 states that different methods might be appropriate for different applications and recommends that multiple methods and sensitivity studies be considered when selecting an aggregation method.

In its letter dated October 20, 2016, the licensee provided estimates of the risk attributable to debris. The licensee presented risk results using the arithmetic mean and the geometric mean to allow comparison. Per NUREG-1829, providing analysis results under differing assumptions helps identify the sensitivity of the results to those assumptions. The NRC staff reviewed the licensee's information and concludes that the sensitivity analysis of the risk results to the choice of aggregation method is an acceptable way to address this source of uncertainty because it is consistent with the recommendation in NUREG-1829.

The NRC staff reviewed the licensee's information on initiating events and concludes that the initiating event frequencies selected by the licensee are acceptable because:

- They were obtained from NUREG-1829, which is considered to be the most current set of values available.
- The licensee interpreted the NUREG-1829 data in a manner consistent with the guidance in NUREG-1829.
- The licensee provided an analysis demonstrating that other location-specific LOCA contributors were negligible when compared to the frequency of LOCAs caused by long-term material degradation.
- The licensee performed sensitivity analyses to address the selection of LOCA frequencies from NUREG-1829 using the arithmetic mean and the geometric mean.

3.4.2.3 Failure Mode Identification

The following are potential debris-related failure modes for the ECCS long-term core cooling function. The NRC staff expects each of these failure modes to be considered and specifically evaluated or shown to be irrelevant to the risk-informed evaluation. Other potential failure modes are to be evaluated, as necessary, for plant-specific conditions. The licensee evaluated each of the phenomena below and did not identify additional failure modes. These failure modes are only those related to debris.

a. Excessive head loss at the strainer leads to loss of net positive suction head (NPSH) margin for adequate operation of the pumps;

b. Excessive head loss at the strainer causes mechanical collapse of the strainer;

c. Excessive head loss at the strainer lowers the fluid pressure, causing release of dissolved gases (i.e., degassing) and void fractions in excess of pump limits. Vortexing and flashing may also cause pump failure;

d. Debris in the system downstream of the strainer exceeds ex-vessel limits (e.g., blocks small passages in downstream components or causes excessive wear);

e. Debris results in core blockage and decay heat is not adequately removed from the fuel;

f. Debris buildup on cladding results in inadequate decay heat removal;

g. Debris buildup in the vessel leads to potential excessive boron concentrations within the core; and

h. Debris prevents adequate flow to the strainer or prevents the strainer from attaining adequate submergence.

The NRC staff evaluated the licensee's analysis and determined that the failure modes evaluated by the licensee include all those that could reasonably lead to debris induced failure of long-term core cooling. Therefore, the NRC staff concludes that the licensee included the appropriate failure modes and evaluated them acceptably.

3.4.2.4 Changes to the Base PRA Model

As discussed in SE Section 3.4.1, "NRC Review of the Base PRA Model," the licensee's simplified, bounding approach to estimating the risk attributable to debris used the licensee's base PRA model in a limited manner. No base PRA model changes were necessary in order to estimate the risk attributable to debris using this simplified method. Since no changes were made to the base PRA model, the NRC staff did not perform a detailed evaluation of this area. Consistent with the guidance in RG 1.174, which states that risk-informed applications should include the effects of past applications, the NRC staff expects that the base PRA used for future submittals will include consideration of the risk attributable to debris unless it can be shown to not affect the decision being made. This aligns with the overall guidance in RG 1.174 and RG 1.200, both of which state that the PRA should realistically model the as-built, as-operated plant and with the ASME PRA Standard,[64] which explicitly states that phenomenological conditions (e.g., effect of debris on NPSH) should be included in accident sequence or system models.

3.4.2.5 *Scenario Development*

For the purposes of this SE, the term "scenario" means an initiating event followed by a plant response, such as a combination of equipment successes, failures, and human actions, leading to a specified end state, such as event prevention, core damage, or large early release.

As described above, the only scenarios of interest for determining the risk attributable to debris are LBLOCA events occurring with the reactor at full power. A typical scenario is a pipe break LOCA at a weld location that results in debris generation and transport to the strainer and core. If a detailed analysis of each scenario is performed, various break sizes at each location can be analyzed. Debris generation and transport specific to each scenario can be determined, considering variations in parameters such as pool temperature, chemical effects, operator response, etc. However, the licensee employed a simplified approach that assumed that core damage occurred any time the scenario resulted in a debris amount greater than the threshold determined by testing. This approach simplified the number of different scenarios to be considered.

In Attachment 1-3 of its letter dated October 20, 2016,[30] the licensee identified two different scenarios to be considered in its simplified approach. Specifically, the licensee stated that the impact of debris on ECCS and CSS performance depends on the plant configuration at the time of recirculation. The licensee specifically analyzed two cases:

- Case 1 – two or three ECCS and CSS trains available
- Case 2 – one ECCS and CSS train available

The licensee's justification for limiting the analysis to these two plant operating conditions is stated as follows:

> The ECCS debris screen testing considers two trains operating, one train idle, to be consistent with deterministic design assumptions that is, one train is assumed failed. The more likely case (not tested) would be three trains in operation in which case the debris would be spread over three ECCS strainers. It is clear that the three train case is bounded (in terms of debris loading) by the two-train case.
>
> The limiting single ECCS/CSS train operation is when all pumps are operating on one ECCS strainer. For example, if one LHSI [low head safety injection] is operating on one strainer and one HHSI [high head safety injection] is operating on a different strainer, the maximum loading on each strainer would be less than if both pumps operated on the same ECCS strainer. If the CSS were to be operating on the third strainer, approximately one third of the debris would load on the that [sic] (third) strainer without passing to the RCS. However, in a risk-based assessment, single train operation is possible and for certain scenarios, single train operation is assessed to go to success in the PRA. In the STP ECCS design, single train operation would result in twice the debris load on the operating strainer. Therefore, the breaks that could be tolerated would be those with one half the tested (two-train operation) debris load.

These two cases were the only scenario differences identified by the licensee as important to the risk assessment.

The bounding approach used by STP for strainer head loss did not explicitly model time-dependent effects (such as chemical precipitation) but used a bounding debris load to envelope long-term effects. Details on the evaluation of the assumed debris load and test results are included in SE Sections 3.4.2.8, "Impact of Debris," and 3.4.2.8.3, "Head Loss and Vortexing."

It should also be noted that for the evaluation of the amount of debris that may penetrate the strainer and reach downstream components, including the reactor vessel, the licensee used different assumptions for plant equipment states that are conservative with respect to the downstream debris amounts. The methodology used for calculating the debris penetration amounts is evaluated in the SE Sections 3.4.2.6, "Debris Source Term," and 3.4.2.7, "Debris Transport."

The NRC staff evaluated the licensee's information on the two cases discussed above and concludes, as discussed above, that the licensee adequately characterized the important scenario differences for the simplified, bounding approach to estimate the risk attributable to debris.

3.4.2.6 Debris Source Term

The Debris Source Term section describes the debris that may be generated during an initiating event and evaluates whether this debris can reach the ECCS strainers.

The following review areas are evaluated below in SE Sections 3.4.2.6.1 through 3.4.2.6.6:

- Break Selection
- Debris Generation and Zone of Influence
- Debris Characteristics
- Latent Debris
- Coatings
- Containment Material Control

For the ECCS strainer portion of the evaluation, the licensee conducted two debris source term evaluations. The first debris source term was calculated as an input to a strainer head loss test that was conducted in July 2008. Prototypical head loss testing was conducted based on this debris load. The NRC staff found the licensee's assumption on the amount of fibrous debris that could be generated during a LOCA underestimated debris loads expected by the NRC staff. Therefore, the test was determined to be non-conservative. For the second debris source term for individual break locations, the licensee used many calculations to determine which breaks could generate more fibrous debris than was included in the July 2008 test. Any break that generates and transports more debris than that represented in the test is considered to fail deterministic criterion and is evaluated using risk. This SE section evaluates these parts of the deterministic analyses to determine whether the licensee used appropriate inputs to the risk-informed analysis.

In Attachment 1-2 of its letter dated August 20, 2015,[20] the licensee stated that the particulate quantities added to the July 2008 head loss testing were a combination of the highest debris destruction quantities from break locations for each individual particulate source. For instance, the tested qualified coatings, Marinite®, and Microtherm® quantities each came from separate break locations that maximized their individual destruction amounts. The licensee stated that after the July 2008 testing, all Marinite® was removed from the containment. Although the Marinite® was removed from containment, the licensee later used some of the Marinite® debris that was included in the test to account for coating debris that was not fully accounted for in the July 2008 test. The amounts of debris that may reach the sump strainer and the amounts that were included in the test are discussed in the Debris Generation (SE Section 3.4.2.6.2) and Head Loss and Vortexing (SE Section 3.4.2.8.3) sections.

The licensee initially conducted evaluations to resolve GL 2004-02 using NRC guidance (NEI 04-07 [48]) that provides deterministic methods to demonstrate that the ECCS and CSS operate adequately considering debris effects. The licensee later determined that it would use a risk-informed methodology to resolve GL 2004-02. The acceptance criteria for strainer performance was derived from a head loss test performed by the licensee in July 2008. After the licensee changed from a deterministic method to a risk-informed method, the assumptions and inputs to the test were rendered relatively unimportant. Therefore, a portion of the information the licensee previously provided is no longer applicable to the NRC staff's evaluation and is not discussed in this SE.

The important information was the list of the amount of each material included in the testing and an evaluation that determined whether the testing was conducted under realistic or conservative conditions when compared to the plant. Evaluations of the amount of debris that may enter the RCS under various conditions are also important for the risk-informed evaluation.

The NRC staff's review focused on the licensee's evaluation of whether the acceptance criterion were developed using adequate methods. For example, the NRC staff did not review the licensee's calculation of the debris amount included in the 2008 strainer test, since it is not relevant to this review. However, how the licensee calculated the amount of debris that can be produced by each potential scenario and transported to the strainer, is relevant.

3.4.2.6.1 Break Selection

When determining the amounts of debris to include in the strainer head loss tests, the licensee followed the break selection criteria specified in RG 1.82, Revision 4.[56] The licensee's 2008 test debris quantities were based on pipe break locations that maximized debris quantities in the debris generation analysis. For the risk analysis, the licensee evaluated the potential for debris generation from all initiating events that were screened into the analysis.

The NRC staff determined that the licensee selected break locations that would maximize the amount of debris arriving at the strainer when defining inputs for the 2008 test. The original locations were at the 29-inch hot-leg inside the steam generator compartment and the secondary shield wall, the 29-inch RCS hot-leg at a nozzle in the reactor cavity, and the 31-inch RCS crossover line. The use of these breaks as the basis for the debris inputs to the 2008 strainer head loss test is appropriate considering the assumptions used by the licensee during testing.

The NRC staff notes that the licensee did not analyze ASME Class 2 piping for sump performance for the 2008 test inputs because Class 2 piping systems do not affect sump performance. The staff found that the licensee adequately justified this approach, but it had to be revisited for the risk-informed analysis because some beyond design basis events involving

Class 2 piping may lead to sump recirculation in the feed and bleed mode. The licensee screened out these secondary breaks that could lead to sump recirculation (feed and bleed) from the risk-informed analysis as having insignificant effect on plant risk.

The licensee also screened out other potential initiating events like seismically induced LOCA, water hammer-induced LOCA, and secondary side breaks from the risk-informed analysis based on their very low risk contribution, which is discussed in SE Section 3.4.2.1, "Scope of the Systematic Risk Assessment."

For the deterministic testing, the NRC staff concludes that the licensee selected the appropriate pipe break locations and sizes that generate the most challenging debris amounts when preparing for the July 2008 strainer head loss test. The NRC staff concludes that the licensee analyzed the impact of debris generation on the sump performance for all scenarios and break locations to support its risk-informed analysis.

For the risk-informed evaluation, the NRC staff concludes that the licensee's break selection evaluation is acceptable because STPNOC evaluated all ASME Code Class 1 weld locations and considered and screened out other potential break locations. Although the deterministic guidance states that the licensee should evaluate all RCS pipe locations for potential rupture, the NRC staff concluded that, for the risk-informed evaluation, the evaluation of piping only at welds is acceptable. The NRC conclusion is based on industry operating experience that states that piping is much more likely to fail at welds; and even at welds, the likelihood of a rupture that would affect sump recirculation is small. The NRC staff considered that branch connections and elbows may be more highly stressed than other piping sections. The NRC staff concluded that each of these piping components have welds in close proximity. Because each weld is evaluated for jets that may emanate in all directions resulting in zones of influence (ZOI) in all directions radial to the pipe for partial breaks, and for spherical jets from double-ended guillotine breaks (DEGBs), breaks in the elbows or branch connections are well covered from a debris generation perspective. The large ZOIs associated with debris generation combined with the close proximity of the weld locations, make it unlikely that debris amounts from breaks at elbows and branch connections would be significantly larger than those from nearby weld locations. The NRC staff also determined that the potential for longitudinal breaks in piping is very small and need not be considered because the licensee does not have longitudinal pipe welds in the RCS.

3.4.2.6.2 Debris Generation and Zone of Influence

The licensee conducted a debris generation analysis to calculate the amounts of each type and size of debris that should be added to the July 2008 strainer test. The licensee stated that the debris generation analysis for the 2008 testing considers each ZOI to be defined based on the material destruction pressures. The licensee stated that refinements in the STPNOC analysis include debris-specific and non-spherical ZOIs. These refinements were used in both the 2008 analysis and the risk-informed analysis. The destruction pressures and associated ZOI radii for the particulate-based insulating materials in STP containment are listed below. These assumptions were also carried forward to the risk-informed debris generation calculations.

As a background for the following discussion, L/D is the terminology used to define the damage zone for various materials; L is the radius of the spherical or hemispherical jet, and D is the diameter of the break. For DEGBs, D is equal to the inner diameter of the pipe and a spherical jet is assumed. For single-sided breaks, a hemispherical jet is assumed. More robust materials have higher damage pressures and smaller L/D values.

The licensee stated that NEI 04-07 does not recommend a destruction pressure or ZOI for Marinite® and insufficient data exists on its material properties and destruction pressure. Since the insulation is covered with 3/16-inch stainless steel plates, the destruction pressure was assumed to be equivalent to reflective metal insulation, which is 114 pounds per square-inch (psi) gauge (psig), which corresponds to a 2.0D ZOI. The licensee stated that although Marinite® was used in the 2008 head loss testing, all Marinite® has since been removed from containment and replaced by NUKON™ as part of a plant upgrade in 2009. Debris generation for this material was not required in the risk-informed evaluation since it had been removed from containment.

The licensee stated that the material specifications for Microtherm® were insufficient to determine an appropriate destruction pressure and ZOI, so the lowest destruction pressure (Min-K at 2.4 psi) and the greatest ZOI (also Min-K at L/D of 28.6) identified in NEI 04-07[48] were used in the STP analysis. The ZOI of 28.6 was used for Microtherm® in the risk-informed analysis as well as the original analysis.

For the 2008 testing, the licensee assumed that the ZOI for fibrous insulation was 7D. The amount of fiber in the test, based on the 7D ZOI, established the acceptance criteria used by the licensee in the RoverD evaluation. The fibrous debris loading for the 2008 test is not consistent with staff guidance and the NRC staff's evaluation of the resulting impact is discussed further below. Additionally, the NRC staff's debris generation evaluation is discussed below.

For the risk-informed analysis, the licensee assumed a destruction pressure of 6 psi for both NUKON™ and Thermal-Wrap™ fibrous insulation. This corresponds to a ZOI of 17D and is consistent with NRC staff guidance. The amount of fiber in the test was much less than the largest potential amount that can be generated from the limiting break.

For the risk-informed evaluation, the licensee assumed that top-coated inorganic zinc (IOZ) with a qualified epoxy coating system had a ZOI of 4D. For untopcoated IOZ, a ZOI of 10D was assumed. The licensee also reported results assuming a ZOI of 10D for both top-coated and untopcoated qualified IOZ.

The licensee stated that robust barriers can be credited to prevent further expansion of the break jet. The volume of a spherical ZOI with a radial dimension extending beyond barriers is truncated at the barrier. NEI 04-07 stipulates that deflection/reflection need not be considered, but "shadow" surfaces of components should be included in the analysis. The licensee stated that to avoid complications from equipment shadowing, only concrete structures were credited as robust barriers.

For the risk-informed evaluation, the licensee originally intended to show that the amounts of particulates included in the test bounded the amounts that could arrive at the strainer in the plant. Based on this assumption, the licensee planned to perform debris generation for only fibrous debris amounts. It was determined that some particulate debris types may not have been bounded in the testing. Therefore, the licensee calculated fibrous debris amounts for all breaks and performed validations to show particulate debris amounts for all potential break locations were bounded by the 2008 test.

The licensee's computational efforts focused on estimating amounts of fibrous debris and IOZ coatings. The licensee concluded that other debris types were bounded by the debris included in the 2008 test. Accordingly, the focus of the NRC staff evaluation in this section is on fiber and IOZ debris. Microtherm® insulation is a problematic debris type that is also a focus of the NRC staff's evaluation.

The licensee identified a total of 628 welds located on pipes distributed throughout the reactor containment building as potential LOCA break locations. The licensee developed a CAD model to identify the location and geometry of welds, insulation, coatings, and robust barriers such as concrete structures. The licensee built the CAD model based on documents such as piping isometric diagrams, insulation and equipment component specifications, power plant blueprints, and direct measurements. The licensee used the reactor containment building CAD model as a tool to quantify the amount of fibrous debris from NUKON™ and Thermal-Wrap™ and the amount of qualified IOZ coating debris that could be produced from a postulated LOCA event at each weld location. The licensee also determined the amount of Microtherm® insulation debris that could be generated by LOCA jets from these welds.

The licensee also considered the latent debris contribution to fibrous and particulate debris amounts. The licensee considered that it was not appropriate to credit unqualified coatings as remaining in place in post-LOCA conditions, and assumed unqualified coatings as debris. The amounts of latent and unqualified coating debris were modeled by the licensee as independent of the size of the postulated LOCA break.

The licensee determined the smallest size break at each location that would result in debris deposited on the strainer that exceeded the acceptance criteria determined by the July 2008 test. The smallest break at each location was called the critical break size. The size of a jet from a pipe break is calculated based on the area of the opening in the break. In case of a full pipe break, the licensee used a spherical ZOI based on the material L/D using the applicable pipe inner diameter. The licensee stated that each scenario-specific break is assumed to have either a spherical ZOI for a DEGB or a hemispherical ZOI for partial breaks. A DEGB was assumed at each potential break location. If the debris generated by the DEGB was less than the acceptance criteria determined by the test, no further evaluation was performed. For breaks that generated more than the acceptance criteria, the break size was reduced and the break was considered to be a partial break. For the partial breaks, a hemispherical ZOI was assumed with a direction normal to the pipe axis. In determining the smallest break that would just fail the acceptance criteria at each location, the licensee used a systematic sampling methodology with a break size resolution of 0.01 inches and an angular resolution of one degree. The break size was reduced in 0.01-inch increments and the direction was swept in a full 360 degree arc around the pipe until a break size was identified that did not exceed the debris acceptance criterion. In Attachment 1-3 of its letter dated August 20, 2015,[20] the licensee clarified this by noting:

> In other words, for each critical weld location, we sampled 360 jet directions at a break size 0.01 inches smaller than the reported smallest break size and did not find a break that exceeded the threshold. By utilizing this systematic sampling method with high break size and angular resolution, we know that we have found the smallest break that fails at each of the critical weld locations to within a 0.01 inch tolerance.

The licensee programmed algorithms in the code, called CASA Grande, to automate computation of debris amounts generated by each postulated break location. Information in the CAD model was exported into files that were used as input to the CASA Grande code. The information exported includes the geometry of pipes, equipment, concrete walls, insulation distribution, and weld placement. The CASA Grande code was used to digitally draw ZOIs shaped as spheres (for DEGBs) or hemispheres (for partial pipe breaks) centered on all of the 628 welds that were identified as potential locations for LOCA events. The licensee considered robust barriers, such as concrete walls, to limit the ZOI. The licensee quantified the amount of insulation and qualified coatings within the ZOI, and unprotected by robust barriers, to estimate the amount of debris in case of a postulated LOCA. The licensee did not consider steel equipment (e.g., steam generators, pressurizers, and reactor coolant pumps) to be robust barriers. In other words, insulation and coatings were considered damaged if within the ZOI, even if located in the "shadow" of steel equipment. Steel equipment was modeled as transparent to LOCA break jets for the sake of estimating debris amounts. Only concrete structures were modeled as robust barriers.

For all debris types except fiber and IOZ, the tested amount of debris was greater than that predicted to be generated by breaks at the evaluated weld locations. The licensee was able to show that even though the amount of IOZ in the test was not bounding, the total amount of particulate in the test was greater than the total particulate from any break. This is discussed in more detail below. Because fiber was the only debris component not bounded, it was assigned as the acceptance criterion for the strainer evaluation.

The licensee initially determined 3.972 cubic feet (ft^3) of Marinite® could be generated by LOCAs in its debris generation evaluation and later removed all Marinite® insulation from the containment and replaced it with NUKON™.

The licensee's original debris generation evaluation determined that 2.2 ft^3 of Microtherm® could be generated by LOCAs. The licensee's test accounted for 2.2 ft^3 of Microtherm®, but the licensee later determined that only 0.959 ft^3 of Microtherm® could be generated by any single break.

For low-density fiber glass (e.g., NUKON™, and Thermal-Wrap™), the licensee defined three subzones within the ZOI as destruction zones to estimate the debris size distribution (amounts of fiber fines, small fiber, large fiber, and intact fiber blankets). Closer to the break, the debris produced is mainly fines and small fiber. Further from the break, the debris is mostly large pieces of fiber. Some insulation is considered to remain intact (within its protective cover) if far from the break, even if within the ZOI. The technical basis of this debris size distribution approach is an NRC audit report dated July 29, 2008,[65] for Indian Point Nuclear Generating Units Nos. 2 and 3, based on industry testing. The report has been used by several licensees to refine fibrous debris generation amounts.

The licensee considered constant amounts of latent debris and unqualified coatings equal to 30 pounds-mass (lbm) of fiber fines, 170 lbm of latent particulate, and 369 lbm of unqualified coatings.

Two cases were considered for debris loading of the strainers: one case for two ECCS trains running and one case for a single ECCS train running. The single case results in more failures because the strainer area is one-half of the two-train case so only one-half the amount of debris is required to block the strainer.

On December 23, 2009, the NRC issued a request for additional information (RAI) to the licensee.[66] The NRC staff requested that the licensee justify that a ZOI of 7D was appropriate for NUKON™. Since the licensee reverted from a 7D to a 17D ZOI for the risk evaluation and because 17D is the approved ZOI per NEI 04-07, these questions are no longer applicable and no response is needed. In question 11, the NRC staff requested the licensee to provide justification for the ZOI used for Marinite®. Since the licensee removed all the Marinite® from the containment, this question also is no longer applicable and no response is needed.

In question SSIB-3-1, dated April 11, 2016,[67] the NRC staff requested that the licensee provide information regarding the treatment of tags and labels with respect to debris generation. In its response dated May 11, 2016,[22] the licensee stated that the tags and labels installed in the STP containments were impervious to post-LOCA conditions and had been tested and shown not to transport to the strainer. Therefore, the licensee did not include them in the strainer evaluation or testing. The NRC staff found this information responsive to the question.

The NRC staff evaluated information in the RoverD supplement (Attachment 1-3) of the licensee's letter dated August 20, 2015,[20] and descriptions of the reactor containment building CAD model provided in Enclosure 4-3, "Engineering (CASA Grande) Analysis," of the licensee's letter dated November 13, 2013.[4] In addition, the NRC staff participated in a technical audit on May 12-13, 2015, to inspect key components of the RoverD analysis, including the CAD model.[68]

In SSIB-3-4, the NRC staff questioned whether the amounts of particulate debris used in the 2008 strainer head loss testing were bounding of the quantities calculated to reach the strainer in the risk-informed evaluation. The licensee provided information to show that the test amounts were bounding. This information is discussed in more detail in SE Section 3.4.2.8.3, "Head Loss and Vortexing."

The licensee's response shows that the 2008 strainer head loss test debris load bounded the largest amount of particulate that is predicted to be generated from any LOCA break within the STP containments. The licensee used Marinite® in the 2008 test, but subsequently removed all Marinite® from both containment structures at STP. In the RoverD analysis, the licensee used Marinite® as a replacement for coatings that were not included in the test since the total particulate test amounts bounded the particulate mass of all breaks evaluated in the RoverD analysis. Marinite® is known to generate less dense debris than coatings debris, similar to Cal-Sil, such that it takes up more volume in the debris bed, which results in the potential for higher head losses. Therefore, the substitution of Marinite® for coatings is conservative.

The NRC staff finds that the licensee properly quantified amounts of debris that could be generated within the STP containments. For the purpose of the STP evaluation, some debris types were considered to be bounding values and the amounts were the same for all breaks. The CAD model included detailed information necessary to calculate amounts of low-density fiber glass, Microtherm®, and IOZ coatings within any break ZOI. The licensee computed debris amounts using CASA Grande. The CASA Grande code used CAD information to determine debris amounts for each break size of interest. The CAD model clipped the ZOI to account for robust barriers. The licensee also manually estimated the debris amounts from the CAD model directly and compared these results to the corresponding CASA Grande output. There was satisfactory agreement between the two methods. The licensee refined the discretization of components in the CASA Grande code until the debris amounts were in adequate agreement with the CAD model computations. The CASA Grande code outputs total debris amounts of low-density fiber glass, IOZ, and Microtherm® that are compared to pre-computed amounts using the CAD model, every time the model is executed. The NRC staff concludes that algorithms in the CASA Grande code for the computation of debris were properly implemented and verified by the licensee. The licensee adequately considered random factors such as the jet orientation, and identified maximal debris amounts to compare to strainer tests. The implementation of the random jet orientations and connection and significance to computations of the CDF are evaluated in the SE Section 3.4.2.11, "Sensitivity and Uncertainty Analysis." The NRC staff concludes that the licensee's scheme to define critical break sizes to determine the amount of debris generated and transported to the strainer is adequate to provide inputs to the CDF calculations.

The NRC staff finds that the licensee adopted guidelines in the NEI 04-07 report to (i) define ZOIs; (ii) account for robust barriers; (iii) compute debris amounts of low-density fiber glass, Microtherm®, and IOZ; and (iv) estimate debris amounts associated with latent fiber and particulate, and unqualified coatings. The NRC staff finds the approach to compute the low-density fiber glass size distribution as a function of proximity to the break acceptable. The fiber size distribution methodology was considered equivalent to accepted methodologies. A similar methodology was used by Indian Point and reviewed by the NRC staff as part of an audit of its GL 2004-02 closure. The size distributions used for each sub-ZOI are listed in Table 3.2-3 of the Indian Point audit report. [65]

The NRC staff concludes that substituting Marinite® as a surrogate for coatings debris on a mass basis is acceptable based on two factors. First, Marinite® is known to be a problematic debris type, very similar to Calcium Silicate (Cal-Sil). Second, Marinite® is less dense than coatings debris so that it takes up more volume in the debris bed resulting in the potential for higher head losses. This issue is further addressed in SE Section 3.4.2.8.3, "Head Loss and Vortexing."

The NRC staff used verifications performed by a contractor, Southwest Research Institute®, Center for Nuclear Waste Regulatory Analyses, to validate that the licensee's calculations were performed accurately and used acceptable assumptions. The contractor used a combination of confirmatory calculations, engineering review, and exercising of the licensee's software to perform the verifications. The contractor's review allows the NRC staff to conclude, with a high level of confidence, that the calculations for debris generation were conducted and applied properly.

The NRC staff reviewed the licensee's evaluation against the NRC staff-accepted guidance and concludes that the licensee has adequately determined for each postulated break location, the zone within which the force of the jet emanating from the break would be sufficient to generate debris and the amount of debris that would be generated. The licensee determined bounding values of all types of debris except fibrous and used these values to specify debris amounts for the 2008 strainer head loss test. Because the head loss test did not include a bounding amount of fibrous debris, the licensee performed location-specific evaluations for fiber. The licensee's methods are consistent with staff guidance. Therefore, the NRC staff concludes that the licensee's evaluation of the ZOI and debris generation is acceptable. The critical break sizes and amounts of debris from each of these were determined appropriately. The correlation between debris amounts and CDF are evaluated in SE Section 3.4.2.10, "Systematic Risk Assessment."

3.4.2.6.3 Debris Characteristics

The licensee included the following debris types in the scope for its evaluation of debris characteristics: NUKON™, Thermal-Wrap™, Microtherm®, and Marinite®. While the licensee included Marinite® in its July 2008 test data, the Marinite® was removed from Units 1 and 2 in 2009. The licensee did not include reflective metal insulation in the scope because it did not consider it transportable. The licensee assumed 100 percent fines for those materials for which debris characteristics had not been well defined, such as Marinite® and Microtherm®. The licensee used the following as-fabricated densities, microscopic densities, and dimensions for the debris types at STP:

Debris Material	As-Fabricated Density (lbm/ft^3)	Microscopic Density (lbm/ft^3)	Characteristic Diameter (μm)
Fibrous (Fine) Material Characteristics			
NUKON™	2.4	175	7
Thermal-Wrap™	2.4	159	5.5
Particulate Debris Characteristics			
Microtherm®	15	187	2.5 to 20

The licensee used fiber, Marinite®, and Microtherm® in the 2008 head loss testing. The licensee did not attempt to use the microscopic debris characteristics to calculate the head loss behavior of the debris. Therefore, the major debris physical characteristic important for head loss is the fiber density, which is used in the calculation of the mass of fiber arriving at the strainer. The other important characteristic is the size distribution of the fiber after being damaged by a LOCA jet. The sizing is discussed below. In the 2008 head loss testing, the licensee used the actual debris types. Microtherm® and Marinite® (which has been removed from the plant) were added in powdered form. Latent debris was added as fine fibers and an appropriate size distribution of particulates. Coatings characteristics are discussed in SE Section 3.4.2.6.5, "Coatings."

For its risk-informed analysis, the licensee used a 17D ZOI for Thermal-Wrap™ and NUKON™, which is consistent with NEI 04-07 guidance. The licensee then further analyzed the 17D ZOI using Alion[‡] proprietary subzones, which defined different percentages of debris sizes for each subzone; the debris sizes consisted of fines, small pieces, large pieces, and intact blankets. The licensee used a similar approach to that used by Indian Point Generating Station, which had been reviewed and found acceptable by the NRC staff.[65]

On December 23, 2009, the NRC staff issued an RAI to the licensee[66] requesting that the licensee to explain its treatment of Thermal-Wrap™ within a 5D ZOI subzone. The licensee responded that it had assumed a debris size distribution within the 5D ZOI subzone for this material that was non-conservative compared to staff guidance. The licensee later changed its assumption to align with staff guidance, effectively answering the staff concern.

The SSIB RAI 13 requested the licensee to clarify how the small fines category was split between fines and small pieces, and how the split between fines and small pieces was implemented when preparing debris for the head loss test. The licensee responded in a letter dated August 20, 2015,[20] stating that it generated 30 percent of the small fines as fine fiber debris for the 2008 test. The NRC staff reviewed the response and noted that the licensee assumed 30 percent of the small fines (which are 60 percent of the total) were fine, which is an overall fraction of 18 percent fines. The NRC staff notes that this fraction of fines would be considered low for total fine fiber percentage when erosion is included when calculating the amount of debris that transports to the strainer. However, since the licensee used fine fiber as a metric for success by comparing the amount in the 2008 test to the amount predicted by RoverD to reach the strainer, the NRC staff finds this percentage is acceptable as applied by the

‡ Alion Science and Technology is one of STPNOC's contractors for the GSI-191 effort.

licensee because it was only used as a test input. The calculations for amounts used in the RoverD evaluation used staff-approved guidance.

The NRC staff reviewed the licensee's evaluation of debris characteristics and determined that the information provided is consistent with the NEI 04-07 guidance and the associated NRC staff SE. The methodology used to determine debris sizes is a more refined method based on NEI 04-07 guidance and is acceptable.

The NRC staff reviewed the licensee's evaluation against the NRC staff-accepted guidance and concludes that the licensee appropriately characterized the debris for use in determining the transportability of debris and its contribution to sump strainer head loss. In addition, the debris surrogates used in the testing appropriately represented the debris that can be generated in the plant. Therefore, the NRC staff concludes that the licensee's evaluation of debris characteristics is acceptable because it is consistent with the NRC's SE related to NEI 04-07.

3.4.2.6.4 Latent Debris

The licensee evaluated latent debris by completing walkdowns for Units 1 and 2 per NEI 02-01, Revision 1, "Condition Assessment Guidelines Debris Sources Inside PWR Containments," September 2002.[69] The licensee evaluated the quantity and composition of the latent debris by sampling Unit 1 for latent debris per NEI 04-07. Because Units 1 and 2 are similar designs and material compositions, and similar maintenance practices are employed in each, the licensee only sampled Unit 1 and applied its values to those of Unit 2. The total calculated values for latent debris based on the sampling program was less than 160 pound-mass (lbm) for each containment. The licensee conservatively assumed 200 lbm of latent debris in each containment. The licensee assumed 85 percent particulate and 15 percent fiber mix for the latent debris. Below are the values used for the latent debris source term in the risk-informed analysis, based on the assumptions of 200 lbm and a 15 percent fiber to 85 percent particulate ratio.

Latent Debris Source Term			
Latent Debris Type	**Mass (lbm)**	**Density (lbm/ft^3)**	**Characteristic Size (ft)**
Dirt and Dust	170	169	5.67 E-05
Latent Fiber	30	175	2.3 E-05

Although the microscopic fiber density for latent fiber is listed at 175 lbm/ft^3, the licensee used 2.4 lbm/ft^3 as the macroscopic density. This is the same density acceptable for NUKON™ and other low-density fibrous insulation types.

As stated above, the licensee performed transport testing of miscellaneous debris, such as tags and labels, and determined that these debris types would not transport to the strainer.

The NRC staff reviewed the licensee's response and concludes that the approach is consistent with the guidance specified in NEI 04-07 and the associated NRC SE. The licensee used default values for latent debris amounts even though sampling of the containment found a lesser amount of debris.

The NRC staff reviewed the licensee's evaluation against the NRC staff-accepted guidance in NEI 04-07 and concludes that the licensee has appropriately identified the amounts and types of latent debris existing within the containment so that its potential impact on sump screen head loss can be evaluated. Therefore, the NRC staff concludes that the licensee's evaluation of latent debris is acceptable.

3.4.2.6.5 Coatings

The licensee's protective coatings analysis and assumptions have changed as the licensee's risk-informed approach has evolved. As a result, many potential issues that the NRC staff documented in previous RAIs are no longer applicable to this current evaluation. The applicability matrix, submitted by the licensee on June 9, 2016,[23] documents which RAI responses are no longer applicable and which should still be relied upon by the NRC staff in its SE. The NRC staff's evaluation below of the licensee's response to GL 2004-02 focuses on the current analysis and those RAI responses which remain applicable.

The licensee's original analysis assumed a spherical ZOI of 5D for both epoxy and untopcoated IOZ qualified coatings based on WCAP-16568-P, "Jet Impingement Testing to Determine the Zone of Influence (ZOI) for DBA-Qualified/Acceptable Coatings," June 2006.[70] Subsequent to the issuance of WCAP-16568-P, the NRC staff identified computational errors in the report and issued additional guidance on April 6, 2010,[71] with respect to the ZOI for untopcoated IOZ. The revised NRC staff guidance calls for a 10D ZOI for untopcoated IOZ. The NRC staff requested the licensee to evaluate the discrepancy, and STPNOC performed a sensitivity study to determine the impact that a 10D ZOI would have on the analysis. The licensee's sensitivity study used the CASA Grande code to determine if any scenarios existed in which a 10D ZOI for IOZ would result in a greater quantity of IOZ debris than assumed in the 2008 head loss test. The results of the study are documented in the licensee's submittal dated August 20, 2015.[20] The licensee showed that the 2008 testing, which serves as the basis for the RoverD analysis, bounds all scenarios with an applied ZOI of 10D for IOZ coatings. Therefore, the ZOIs used in the coatings evaluation remain consistent with the existing NRC position as applied to WCAP-16568-P.

The licensee stated that all of the coatings destroyed within the ZOI are assumed to fail as fine particulate. In addition, 100 percent of the unqualified coatings inside containment are assumed to fail as fine particulate. The NRC staff finds that all of these particulate debris loads were appropriately represented in the 2008 strainer head loss testing.

The licensee assumed that the epoxy coatings within the reactor cavity fail as chips. The NRC staff initially questioned this assertion since it is contradictory to staff guidance which has all unqualified epoxy coatings fail as particulate debris. The licensee later clarified that this population of coatings is actually better categorized as degraded, qualified coatings and are, therefore, eligible for treatment as chip-type debris. The treatment of degraded, qualified epoxy coatings as chip-type debris is based on the DBA testing of coatings debris by Comanche Peak Steam Electric Station and is documented in Keeler & Long Report 06-0413.[72] The tests showed that epoxy coatings originally installed as qualified coatings that degraded over time remain more robust that unqualified coatings which fail as fine particulate. This testing was found acceptable for use by licensees in Enclosure 2 of the 2008 NRC staff's review guidance.[52] The staff's review guidance stipulates that licensees must perform a plant-specific analysis of its coatings to take credit for the debris characteristics described in the Keeler & Long Report. In a letter dated October 20, 2016,[30] the licensee provided a description of the coatings in the reactor cavity, a description of their service life, and a justification for their treatment as degraded, qualified coatings. The NRC staff finds that STP justified the treatment of the epoxy coatings in the reactor cavity as degraded, qualified coatings and therefore, as chip-type debris. Thus, the NRC finds the licensee may take credit for reduced transportability of this debris since it meets the criteria of the 2008 NRC staff's review guidance.

For head loss testing, acrylic coating chips were used as a surrogate for epoxy coating chips. It should be noted, however, that the RoverD analysis is based only on the particulate and fiber loading in the 2008 strainer head loss test. Therefore, any head loss associated with the coating chips used in testing is attributed to the fiber and particulate loading in the new RoverD analysis. This is conservative because the chips are larger in size than fiber and particulate, and can have a more significant impact on head loss. Thus, if an equivalent amount of particulate and fiber were used to represent the head loss associated with chips, the threshold value would be greater.

In addition to the acrylic coating chips, pulverized acrylic coating powder was used as a surrogate for particulate debris generated from epoxy coatings, polyamide primer coatings, alkyd coatings, and baked enamel coatings. Tin powder was used as a surrogate for IOZ coatings. All of the surrogate materials are acceptable for testing because they have similar density, size, and shape characteristics to the postulated debris.

The licensee stated in its August 20, 2015, supplement that it periodically conducts condition assessments of the containment coatings as part of the structures monitoring program. Visual inspections are performed to characterize the condition of the coating systems. Areas of degraded coatings are evaluated and scheduled for repair or replacement as necessary. This is consistent with the NRC staff's expectations for a coating assessment program as documented in the 2008 NRC staff's review guidance.

The licensee provided information such that the NRC staff has reasonable assurance that coatings have been addressed conservatively or prototypically. Analysis and testing were performed in a manner consistent with the NRC staff's review guidance. Head loss testing surrogate materials were representative of the size, shape, and density of the actual plant coatings debris. The licensee's coatings assessment program will identify and mitigate any degraded coatings prior to them becoming a debris source, which may challenge the margins in the strainer analysis. Therefore, the NRC staff concludes that the coatings evaluation for STP is acceptable.

3.4.2.6.6 Containment Material Control

The licensee stated that procedures were formulated to monitor, track, control, and reduce latent debris inside containment during normal operation (Modes 1 to 4), outages, prior to containment closeout at the end of the outage, and prior to entry into Mode 4 at the end of the outage. When replacing insulation inside containment, the licensee uses either a like-for-like replacement or handles the modification as a design change that requires approval by STP Engineering.

The licensee uses a design change process that includes programs and procedures to evaluate and control potential sources of debris inside containment. The licensee stated that all design changes will be screened or evaluated per 10 CFR 50.59, "Changes, tests and experiments." The design change process evaluates new insulation materials that differ from the original materials.

On December 23, 2009,[66] the NRC issued an RAI requesting the licensee to provide a more detailed description of the containment foreign material control programs for STP, including references to procedural requirements and a brief description of methods used to maintain cleanliness.

In Attachment 1-5 of its August 20, 2015, supplement, the licensee responded to the RAI. The licensee pointed to item 3.i.1 in Attachment 1-2 of its August 20, 2015, submittal, which stated that STP maintains containment cleanliness during outages by adherence to the housekeeping procedure. The licensee stated cleanliness is emphasized by reactor containment building coordinators and work supervisors and that prior to containment closeout at the end of the outage, coordinators oversee cleanup to ensure no loose debris.

The licensee stated that during normal operation, it uses procedure "Containment Entry and Partial Inspection," to maintain containment cleanliness. The procedure provides for visual inspection of affected areas at completion of each entry when containment integrity is established to verify no loose debris. The licensee stated that work crews are briefed by Operations to emphasize the importance of maintaining containment cleanliness.

The licensee stated that during outages, it maintains cleanliness by adherence to the housekeeping procedure and foreign material exclusion control procedure. The licensee stated that cleanliness is monitored and encouraged by coordinators and work supervisors and that worker training emphasizes containment cleanliness. Outage newsletters, handbooks, signs, site-wide messages, etc., are used to convey expectations of containment cleanliness. In addition, areas of containment are "owned" by certain STP managers to help ensure cleanliness is maintained and the area is properly cleaned at the end of the outage.

The licensee stated that prior to containment closeout at the end of an outage, building coordinators oversee cleanup of the work areas with assistance by dedicated work crews. Potential debris sources are cleaned or removed. The licensee stated that prior to entry into Mode 4 at the end of the outage, Operations performs a surveillance procedure (Initial Containment Inspection to Establish Integrity) to verify containment cleanliness, which contains an extensive checklist detailing all areas of containment that must be inspected for cleanliness prior to plant startup. Visual inspections are also performed by teams typically led by senior reactor operators.

The NRC staff reviewed the licensee's evaluation against the staff-accepted guidance and concludes that the licensee has design and operational measures in place to control or reduce the plant debris source term. Therefore, the staff concludes that the licensee's evaluation of containment material control is acceptable.

NRC Staff Conclusion Regarding Debris Source Term

The NRC staff evaluated each aspect of the debris source term and concludes that the sub-areas of break selection, debris generation and ZOI, debris characteristics, latent debris, coatings, and containment material control were adequately addressed. Based on the evaluations of each of these review areas, the NRC staff concludes that the debris source term evaluation is acceptable.

3.4.2.7 Debris Transport

This section evaluated the licensee's transport calculations used to determine (i) bounding particulate loads for the July 2008 strainer tests and (ii) fiber loads arriving at the strainer for all breaks within the scope of the evaluation.

The licensee's approach to evaluating debris transport was based on NEI 04-07 guidance and the associated NRC SE,[48] and the licensee provided information requested in the Content Guide. The NRC staff also referred to other guidance as referenced below. The licensee used different simplifying assumptions based on the goals for the July 2008 testing contribution and for the updated analysis.

When the licensee performed the 2008 strainer test, transport was considered when calculating the amount of debris that should be included in the test. However, because the test is used to define the acceptable amount of debris that can arrive at the strainer, the implementation of transport for the test calculations does not need to be evaluated. Only the amount of debris included in the test is important for the RoverD analysis. The NRC staff had questions regarding the transport calculations used to provide inputs for the 2008 test, and the transport of debris during the test. Some of these questions, and the licensee's responses, are relevant to the NRC staff conclusions in this SE. These questions are discussed in this section of the SE.

The licensee considered three types of debris sources: (i) debris directly generated by postulated breaks in the RCS, (ii) latent debris already present in the containment structure prior to any break, and (iii) protective coatings used inside the containment that could become debris. The licensee reduced the source debris by estimating the amount that could be trapped in inaccessible locations or settled out during transport to the strainer. The licensee also estimated the amount of debris arriving at the strainer due to erosion of larger debris (assumed to settle prior to reaching the strainer) into fine pieces during transport.

The licensee used debris transport calculations for two separate purposes. The first was to calculate the amount of debris that reaches the strainer from each postulated break location. These debris amounts were used for a risk-informed evaluation of the probability of core damage due to debris-induced strainer failure. Second, the licensee calculated the amount of fiber that could transport downstream of the strainer. Debris that penetrates the strainer can transport to the core and other downstream components and may affect the ability to cool the core.

The licensee used the July 2008 strainer head loss test to evaluate the potential for debris-induced strainer failure under known fiber and particulate debris loads. The licensee used debris transport calculations in the risk-informed analysis to estimate debris loads that would arrive at the strainer for each postulated break location. The licensee used the July 2008 strainer head loss tests to determine a fibrous debris limit. If the amount of fibrous debris calculated by the transport evaluation for a specific break is less than the limit, the effects of a break at that location are considered to be mitigated by the ECCS and CSS. That is, the sump strainers would perform according to design requirements. The risk evaluation assumed that a fiber load in excess of the load included in the 2008 strainer head loss test would cause the strainers to fail.

The licensee described the tested debris loads as bounding for particulate debris (which was found to be acceptable by the NRC staff in Section 3.4.2.6.2), but not for fibrous debris. In the RoverD analysis, the licensee assumed that a fine fiber load on the strainer in excess of the 2008 test amount would result in the strainer failing to achieve the design specifications regardless of the particulate load. The licensee did not calculate particulate debris loads on the strainer in the RoverD analysis, because the 2008 test contained bounding particulate loads. During the review of the RoverD analysis, the NRC staff questioned whether the particulate debris loads included in the July 2008 strainer head loss test were bounding of that which could transport to the strainer. In particular, the NRC was concerned that the unqualified coatings and Microtherm® amounts included in the test might not have been bounding. Based on the information provided in the licensee's October 20, 2016, supplement, the NRC staff determined that the tested amounts of particulate debris were bounding. This issue is discussed in SE Sections 3.4.2.6.2, "Debris Generation and Zone of Influence," and 3.4.2.8.3, "Head Loss and Vortexing."

The licensee based the particulate debris load in the July 2008 testing on the amount of debris transported from bounding source locations for each particulate type. The licensee's approach with respect to the debris sources used for bounding particulate debris are evaluated in the Debris Generation section.

The licensee analyzed the specific effect of each of the four modes of transport for each type of debris generated. The four modes of transport are blowdown (transport of debris by the break jet), washdown (vertical transport by containment sprays and break flow), pool fill-up (horizontal transport to active and inactive areas of the sump pool), and recirculation (horizontal transport in the active portions of the sump pool by recirculation flow). The licensee applied the logic tree approach recommended by NEI 04-07 to determine transport fractions for each type of debris determined from the debris generation calculation. Fines are the most readily transported debris size and were not assumed to settle out during transport. In its August 20, 2015, submittal, the licensee provided the basic methodology used for the transport analysis (see Attachment 1-2, page 20).[20] The licensee also provided a diagram showing the significant parts of the computational fluid dynamics model used to evaluate recirculation transport. The diagram highlights the sump mass sink, various direct and wash spray regions, and the combined break and spray wash regions.

The licensee used Flow-3D® Version 9.0 for the computational fluid dynamics modeling. The key computational fluid dynamics modeling attributes/considerations included computational mesh, modeling of containment spray flows, break flow, and emergency sump flow. Turbulence modeling, steady-state metrics, and debris transport metrics were also included. The licensee also performed a graphical determination of debris transport fractions. The licensee determined the percentage of each type of debris that could be expected to transport through the containment pool to the strainers. The licensee provided plots for each case showing the turbulent kinetic energy and velocity magnitude in the pool, to determine areas where specific types of debris would be transported. The licensee also provided figures and discussions as an example of how the transport analysis was performed for a generic small debris type. This example was illustrative of how all debris types were evaluated with respect to transport.

The licensee stated the following with respect to the transport evaluation:

- Debris interceptors are not integrated into the STP debris transport analysis.

- 98.5 percent of fine debris is transported to the reactor containment building recirculation pool.

Transport logic trees were developed for each size and type of particulate debris generated to determine the total fraction of particulate debris that would reach the sump screen in each of the postulated cases. This information was then used to determine the total amount of particulate used in the July 2008 strainer head loss testing. The postulated cases include (i) a break in the Loop C hot leg and (ii) a break in the reactor cavity, which are considered the two bounding transport cases.

In its submittal dated August 20, 2015,[20] the licensee stated that since fine fiber is the only type of fiber used as a comparison to the fiber quantity in the July 2008 strainer head loss testing, only fine fiber transport fractions need to be calculated for use in the RoverD methodology. Later, because of NRC staff questions, the licensee performed calculations to show that the particulate debris amounts included in the 2008 test were bounding. This is discussed in the Debris Generation and Head Loss and Vortexing sections. The staff found that the analysis considered the appropriate amounts of particulate debris.

The licensee stated[20] that debris erosion is the only area where the debris transport analysis deviates from NEI 04-07 guidance. Where the guidance specifies an erosion fraction of 90 percent for fiberglass debris, the licensee uses less than 10 percent in the recirculation pool and 1 percent for fiberglass debris held up on gratings. The licensee stated that the only insulation debris with the potential for erosion at STP is the unjacketed small and large pieces of NUKON™ and Thermal-Wrap™ fiberglass. The NRC staff developed questions regarding the amount of fiber that could erode from larger pieces in the sump pool. The response to this issue is discussed below.

The licensee stated[20] that tests performed as a part of the drywell debris transport study indicated that the erosion of fiber debris is significantly different for debris directly impacted by containment sprays versus debris directly impacted by break flow. Based on the results of the drywell debris transport study testing, a 1 percent erosion factor was applied for small and large piece fiber debris held up in upper containment, which is consistent with the approach taken for the pilot plant in the NRC staff's SE on NEI 04-07 (Appendix VI).[48] The erosion mechanism for debris in the pool is somewhat different than what was tested in the drywell debris transport study.

The licensee stated that to quantify the recirculation pool erosion fractions for STP, generic 30-day erosion testing was performed.[73] The licensee stated that a statistical verification of the erosion data from a proprietary industry test was performed, which verified that the average erosion fraction used in the logic tree for large and small pieces of fiber in the pool would be less than 10 percent. The NRC staff reviewed and developed conclusions regarding this report that are documented in a letter dated June 30, 2010.[74] The NRC staff concluded that plants that could demonstrate that the testing was conducted under conditions that represented or bounded their plant could assume a 30-day erosion value of 10 percent for fiber settled in the sump pool.

The licensee also calculated the time-dependent rate of fiber passing through the sump strainer for each of the potential LOCA events, based on the potential time-dependent rates of fine fiber arrival at the strainer. The licensee provided the methodology used to calculate the amount of fiber that could pass downstream of the strainer. The licensee used fiber penetration testing to develop a model of fiber penetration through the strainer over time. The testing was performed using a full-sized strainer module under prototypical plant flow conditions. The scaled debris loads during testing were much higher than the fiber amount that is considered to result in a strainer failure. Smaller fiber loads were also tested to verify that they would not result in additional penetration. During testing, the licensee varied the flow velocity through the strainer and the fiber concentration upstream of the strainer to values considered to be bounding of the plant condition. The licensee used the penetration values from the testing to develop the penetration model. The model accounts for flows through the ECCS and CSS and estimates the amount of fiber in each flow path. The model ultimately calculates the amount of fiber that reaches the core on a time-dependent basis. The licensee performed sensitivity studies to determine the potential range of fibrous debris that may pass through the strainer and reach the core.

The licensee provided the amounts of fibrous debris predicted to reach the core following a cold-leg break under various combinations of operating ECCS and CSS pumps on a per fuel assembly basis. The licensee stated that the amounts of fiber that reach the core are calculated to be very low. Under nominal conditions, the loading was calculated to be less than 2 grams per fuel assembly for a cold-leg break. Under more limiting conditions, using beyond design basis pump combinations and the highest penetration values from testing, the licensee calculated that larger fiber loads can reach the strainer. Under the most limiting pump combination and conditions, the licensee calculated that 7 grams per fuel assembly could reach the core following a cold-leg break. The licensee compared the amount of fiber reaching various downstream components, including the fuel assemblies, with safety criteria based on fiber load tolerances. For a hot-leg break, the amount of debris reaching the core has no impact on the licensee's in-vessel evaluation because it assumes that the core inlet is fully blocked in that scenario. For the hot-leg break, the licensee used a thermal-hydraulic analysis to evaluate core cooling considering the effects of debris that penetrates the strainer. The effects of fiber reaching the core are evaluated in SE Section 3.4.3.8, "Impact of Debris."

The NRC staff compared the licensee's debris transport methodology with the guidance provided in NEI 04-07 and the associated staff SE.

The NRC staff used calculations performed by its contractor, Southwest Research Institute (SwRI), to verify that the licensee's calculations were performed accurately and using acceptable assumptions. The SwRI used a combination of confirmatory calculations, engineering review, and the licensee's software to perform the verifications. The results of SwRI's review allow the NRC staff to conclude, with a high level of confidence, that the calculations for transport and debris penetration were conducted and applied properly.

On December 23, 2009, the NRC issued RAIs to the licensee.[66] On August 20, 2015, the licensee responded to the RAIs.[20] For RAIs 14-16, 21, and 23, the licensee stated that RoverD used the fibrous debris amount from the July 2008 test as a datum of comparison to the CASA Grande generated and transported fiber quantities. All break locations where the maximum fine fibrous debris quantities calculated by CASA Grande were below the tested amount are considered to be successfully mitigated by the ECCS and CSS. All locations that result in more fine fibrous debris at the strainer are evaluated using the risk-informed calculations. Because of the licensee's change in the application of the transport analysis conducted for the 2008 strainer test, only those questions that impact the RoverD submittal are discussed below. The previous transport evaluation simply provided the amounts of debris included in the 2008 test and are no longer applicable because the tested amount simply provides the acceptance criteria as a comparison for the RoverD calculations. Only the transport questions related directly to the 2008 test are discussed in this section.

In question 17, the NRC staff asked the licensee to provide the basis for considering a transport case with two sumps operating as the limiting condition for debris transport. The NRC staff reviewed the response and found it reasonable because although the flow velocities caused by three-sump operation would increase and could result in a small increase in transport of debris, this debris would be deposited on three strainers instead of two making the load per strainer less than the two-sump operation case.

In question 19, the NRC staff asked the licensee to estimate the quantity of eroded fines from small pieces of fiberglass debris that would result had the licensee accounted for erosion of the settled debris in the head loss test flume. This issue is important to the RoverD evaluation and is discussed further below in the paragraph on RAI 19 follow-up.

In question 20, the NRC staff asked for justification for the licensee's assumption of 17 percent holdup of latent debris in inactive sump pool volumes. The NRC staff reviewed the licensee's response and found it reasonable because the licensee demonstrated that this holdup would occur. In addition, this issue was addressed in an RAI on the RoverD methodology in which the NRC asked the licensee to provide a comparison of the amount of debris tested and that calculated to arrive at the strainer. This issue is discussed in the SE Section 3.4.2.8.3, "Head Loss and Vortexing."

In question 22, the NRC staff requested that the licensee provide additional information concerning the following debris transport assumptions regarding failed coatings debris: (a) a basis for the zero percent transport fraction for epoxy coating debris inside the reactor cavity for breaks that do not occur within the reactor cavity; and (b) a description of the methodology for determining the transport fraction for failed epoxy coatings outside the reactor cavity, for which transport percentages from 41 to 48 percent were calculated for various scenarios. The NRC staff reviewed the licensee's response and found the logic for transport of the coatings debris reasonable based on the size distribution of the coatings and the fluid velocities in the areas where transport could occur. However, this issue was further explored during the review of the RoverD submittal because of questions regarding the characteristics of the failed coatings.

In question 24, the NRC staff requested that the licensee provide plots of velocity and turbulence contours in the containment pool and other information pertaining to the calculation of flow parameters in the test flume, and compare them to those that were present in the 2008 test. The NRC staff reviewed the licensee's response and found it reasonable as the licensee provided the requested information. The NRC staff's review of the information found the licensee's use of the information in the strainer test acceptable because it was demonstrated that the test adequately represented the plant condition.

In question 25, the NRC staff asked the licensee to discuss any sources of drainage that enter the containment pool near the containment sump strainers, to identify whether the drainage would occur in a dispersed form, and to discuss how these sources of drainage are modeled in the test flume to create a prototypical level of turbulence. The NRC staff reviewed the licensee's response and found it reasonable because the licensee demonstrated that drainage into the pool would not have a significant effect on turbulence near the strainer. Therefore, the test conditions were found acceptable.

In question 26, the NRC staff asked the licensee to identify any debris quantities added to the test flume prior to starting the test pump for the head loss tests, and to provide a technical basis for adding this debris prior to starting the test pump. The NRC staff reviewed the licensee's response and found it reasonable because the addition of a small amount of debris to the test prior to starting the pump was shown to likely have an insignificant effect on the test results. The licensee also performed a sensitivity study to demonstrate that even if the debris that was added to the test prior to pump start did not transport during the test, it would have negligible effect on plant risk. This study was provided as a response to a follow-up question to question 26. Therefore, the licensee's response is acceptable.

In question SSIB-3-5, the NRC staff asked for additional information on why it is acceptable to consider only fine fiber, and not larger pieces, as the acceptance criteria for strainer failure, considering that the amount of particulate debris in the test was bounding. The licensee provided information regarding the methodology used to calculate fiber transport, the test methodology, and observations made during testing that justify the use of fine fiber as the strainer failure metric. The NRC staff finds the information acceptable because it demonstrated that only fine fiber transported to the strainer during the head loss test. Therefore, the NRC staff finds that the licensee's response to question SSIB-3-5 is acceptable.

In question SSIB-3-6, the NRC staff requested that the licensee provide justification for the assumptions used to determine the transport behavior of failed unqualified coatings in the reactor cavity. The licensee provided the requested information which differed depending on whether the break occurred within the reactor cavity or outside the cavity. For breaks inside the cavity, the coatings are more likely to transport to the strainer than for breaks outside the cavity. The NRC staff found that the treatment of the coatings in the reactor cavity was in accordance with staff guidance. Therefore, the NRC staff finds that the licensee's response to question SSIB-3-6 is acceptable.

As a follow-up to question 18, the NRC staff requested justification for the licensee's assumption that the erosion of larger fiber pieces in the pool would be 7 percent. The NRC staff had accepted 10 percent as an acceptable erosion value for plants that could show that industry testing used to justify the value was applicable to their plant-specific conditions. The licensee provided additional information regarding the plant conditions and why 7 percent is an acceptable value. In addition to the licensee provided information, the NRC staff noted that there was opportunity for erosion of larger fiber to occur in the licensee's test and that this erosion was not credited. Although not quantifiable, this provides some additional margin to the erosion term in the test. In addition, the licensee performed a sensitivity study to show that the change in CDF is insignificant between cases that consider 7 percent erosion and 10 percent erosion. This study was presented in response to SSIB follow-up question 19. Therefore, the NRC staff finds that the licensee's response to follow-up question 18 is acceptable.

As a follow-up to question 19, the NRC staff requested that the licensee justify the treatment of small fibrous debris that had settled in the head loss test, but was assumed to transport to the strainer by the transport evaluation. The NRC staff's concern was that the small fiber should have either been on the strainer during the 2008 test as predicted by the transport evaluation, or if it did not actually transport during the test, it should have been included in the calculation for erosion. The licensee provided a sensitivity study that showed that the increase in risk when the erosion of the small fiber pieces was accounted for was insignificant. Therefore, the NRC staff finds that the licensee's response to follow-up question 19 is acceptable.

For SSIB-3-2, SSIB-3-3, and SSIB-3-4, the NRC staff requested additional information regarding the amount of debris that was calculated to transport to the strainer and how that compares to the amount that was included in the 2008 test. The licensee provided a response that justified that the tested amounts of debris, calculated using NRC staff approved methods, bounded the amounts calculated for all breaks. Therefore, the response to these questions is acceptable.

Separate from calculating the amount of debris that transports to the strainer, the licensee calculated the amount of fiber that might penetrate the strainer and affect downstream components. The NRC staff reviewed the penetration test methodology, test results, and the application of the results to the plant. The staff found that the test methodology would conservatively predict the amount of fiber that could penetrate the strainer and that the use of the results as implemented by the licensee was acceptable. For the penetration testing, the licensee varied the test conditions to determine which resulted in the maximum penetration. The licensee used the maximum values from strainer penetration testing for development of their penetration and transport model. The transport model was reviewed in detail by the NRC staff and its contractor, SwRI, during an audit of the RoverD methodology. The licensee performed several cases using the model to determine the amounts of fiber that can reach the core under different plant operating conditions. The values calculated are conservative estimates of what would actually reach the reactor core. In addition, SwRI reviewed the methodology used for predicting the amount of fiber bypass and transport to the core. SwRI created an independent model for the phenomena, including the fiber penetration and transport by the ECCS and CSS, and determined that the RoverD methodology accurately predicts the amounts of fiber that can reach the core. The contractor considered the different scenarios and assumptions used by the licensee and validated that the model was accurate in each case. Therefore, the NRC staff concludes that the licensee's calculations for fiber penetration through the strainer and transport to the reactor core were performed acceptably.

<u>NRC Staff's Conclusion for Debris Transport</u>

The NRC staff reviewed the licensee's transport evaluation against the NRC staff-accepted guidance, and performed confirmatory calculations for debris penetration. The NRC staff concludes that the licensee appropriately estimated the fraction of debris that would transport from debris sources within containment to the ECCS strainers and the amount of fibrous debris that may penetrate the strainers and transport to the reactor core. Therefore, the NRC staff concludes that the licensee's evaluation of debris transport is acceptable.

3.4.2.8 Impact of Debris

This section evaluates the potential effects that the debris, as described in SE Section 3.4.2.6, "Debris Source Term," may have on operation of equipment important to long-term core cooling. The section describes the operation of the ECCS strainers, the ECCS and CSS pumps, and other equipment downstream of the strainer, including the fuel and vessel.

The following review areas are discussed in SE Sections 3.4.2.8.1 through 3.4.2.8.8:

- Upstream Effects
- Screen Modification Package
- Head Loss and Vortexing
- Sump Structural Analysis
- Net Positive Suction Head
- Chemical Effects
- Downstream Effects – Components and Systems
- Downstream Effects – Fuel and Vessel

3.4.2.8.1 Upstream Effects

The licensee's upstream effects evaluation provides a general description of the containment and its subcompartments as well as an examination of each elevation to identify physical and structural features that affect the flow of debris and water to lower containment. The objective is to identify containment choke points and areas of where water could be prevented from reaching the containment sumps. The licensee's evaluation is based on a review of STP design documents including the Updated Final Safety Analysis Report (UFSAR), calculations, and containment drawings.

The licensee stated that the spray/break inventory and debris from upper elevations will eventually flow down to the 19'-0" elevation. The primary flow paths are through grated floor areas at upper elevations. Once at the 19'-0" elevation, concrete flooring routes the flow of water and debris to grated areas inside and outside the secondary shield wall. The primary sources of insulation debris are located above the 19'-0" elevation (e.g., primary RCS piping and components). Therefore, the majority of insulation debris will be trapped at this elevation unless it can fit through standard floor gratings. The licensee judges that this elevation will not become a choke point for flow because, should large debris deposit on floor gratings, the water will pass through multiple parallel grated flow paths to the lower elevations. In addition, there are multiple flow passages between the areas inside and outside the secondary shield wall increasing the grated floor area available to pass flow.

The licensee stated that the recirculation pool forms at the -11'-3" elevation. The ECCS emergency sumps are located in the southern quadrants of containment outside the secondary shield wall. The flow path around the outside of the secondary shield wall is generally open providing large flow passages to the ECCS emergency sumps.

The licensee stated that no measures are necessary to mitigate potential choke points.

The licensee stated that there are only four significant openings through which recirculation water and debris may pass from inside the secondary shield wall to the annular region outside the secondary shield wall at the -11'-3" elevation. These openings are four 30-inch circular vent holes located at a centerline elevation of -8'-6". Since these vent holes are above the floor, the secondary shield wall acts as a curb, or debris barrier, in the flow path to the containment sumps. Only small debris (small enough to fit through standard floor grating) is expected to reach the base floor elevation. Significant accumulation of small debris is not expected to create a dam that would prevent flow through the vent openings. The volume of water inside the shield wall and below the vent holes is considered unavailable to the ECCS emergency sump.

The licensee stated that no new curbs and/or debris interceptors have been installed in response to GL 2004-02.

The licensee stated that the refueling cavity drains via two horizontal 6-inch drains with centerline elevation located 10.75-inch above the bottom of the lower internals storage area. The two horizontal refueling cavity drains have an inside diameter of 6.065-inch and are straight pipe segments approximately 7-ft long. In its letter dated December 11, 2008,[75] the licensee stated that ALION-CAL-STPEGS-2916-006 and Westinghouse letter LTR-CSA-06-45 are the basis for conclusion that the refueling cavity drain will not become plugged with debris. Based on debris generation and transport analyses, it was conservatively determined that 71 ft^3 of fines (individual fibers) and 177 ft^3 of small pieces (less than 6 inches) of fiber insulation may be transported to the refueling cavity. The RoverD analyses assume that the drains do not become blocked by debris, and do not restrict flow from the cavity. No additional water hold-up is assumed for the refueling cavity except that volume required to induce flow through the cavity drains above the cavity floor.

The licensee stated that debris blown out of the steam generator compartments is expected to be distributed evenly around the operating floor (elevation 68'-0"). The refueling cavity drain lines are located on opposite walls of the lower internals storage area and large concentrations of debris are not expected to deposit near both drain lines. There are no drain covers or trash racks covering the drains that would allow fibers to build up and block flow. The largest debris transported to the refueling cavity (less than 6 inches) is smaller than the drain line diameter (6.065-inch). In addition, fibrous debris is not rigid and will deform to fit through the drain if needed. The flow velocity through the drains is greater than the incipient tumbling velocity for 6-inch pieces of NUKON™; however, should debris accumulate in the drain line, the buildup of water behind the debris will provide sufficient driving force to push the debris through the straight pipes.

In an RAI dated December 23, 2009, [66] the NRC staff asked question 44 regarding the potential for large pieces of fibrous debris to be blown to upper containment such that it could transport to the refueling cavity and block the drain lines. This is considered critical because of the large volume of water that may be held up in the refueling cavity. The licensee responded with adequate information such that the NRC staff was able to conclude that large debris would not transport to upper containment. This issue is discussed in more detail in the SE Section 3.4.2.8.5, "Net Positive Suction Head," because of its potential effect on sump level, and therefore the NPSH calculation.

The NRC staff reviewed the licensee's evaluation against the NRC staff-accepted guidance and concludes that, in combination with the discussion in Section 3.4.2.8.5, the licensee has appropriately evaluated the flow paths upstream of the containment sump for holdup of inventory that could reduce flow to the sump and possibly starve the pumps that take suction from the sump. Therefore, the NRC staff concludes that the licensee's evaluation of upstream effects is acceptable.

3.4.2.8.2 Screen Modification Package

The licensee stated that there were no changes to the three independent sump pits as a result of the evaluation of GL 2004-02. The licensee stated that the sump screen above the pit has been removed and now each sump has its own new strainer. The licensee stated that there are no shared components between trains.

The licensee stated that the new strainer assemblies for each of the sumps consist of two 5-module assemblies, one 4-module assembly, and one 6-module assembly. Each module is made up of 11 strainer disks. The licensee explained that the strainer consists of a stainless steel perforated plate with 0.095-inch diameter openings. Flow leaving the strainer assembly enters a four-inlet plenum box (one inlet for each strainer assembly). The licensee stated that the plenum box collects the flow from the strainer assemblies and directs it downward directly into the sump pit. An access cover on the plenum box allows for internal inspection of the sump structures, vortex suppressor, and the strainer assemblies. The licensee stated that the sump pit is now covered with a sump cover plate that prevents material from falling directly into the pit without passing through the strainer assemblies.

The licensee stated that the new strainers have a surface area of 1,818.5 square feet (ft^2) per sump, whereas the old screens had a surface area of 155.4 ft^2 per sump. For the design flow of 7,020 gallons per minute (gpm) per sump, the new strainers have an approach flow velocity of 0.009 ft/sec.

The licensee stated that following installation of the new sump strainers, protective gratings were installed in front of the strainers to preclude inadvertent damage. The framing structure for the protective gratings consists of vertical grating panels attached to metal columns that are welded to base plates that are anchored into the concrete floor. The structure is qualified for Seismic II/I loading to ensure maximum stresses are below the allowable limits. The material is carbon steel, which has a qualified coating applied.

The licensee stated that no piping reroutes were needed for installation of the new sump strainers. The new strainer installation did not require component relocations or additions.

The NRC staff reviewed the design changes and determined that the licensee included appropriate design change information in its submittals in response to GL 2004-02.

3.4.2.8.3 Head Loss and Vortexing

The licensee's initial head loss model calculations relied on correlations to determine the head loss that could occur across the strainer depending on the debris predicted to collect on it. Generally, the NRC has not accepted the use of correlations to predict head loss unless they are bolstered by significant testing conducted under plant-specific conditions. The initial licensee submittal on June 19, 2013, did not provide testing information, so the NRC staff asked questions (RAI dated April 15, 2014) to understand the head loss correlations and their ability to predict head loss. Subsequently, the licensee decided to use a head loss value derived from a plant-specific test to evaluate the operation of the sump strainers rather than using a correlation.

In 2008, the licensee performed a strainer head loss test, with chemicals, that the NRC staff considered to be generally acceptable. The NRC staff asked several questions regarding the test to ensure that it was conducted in accordance with guidance. The RAIs regarding the test are discussed below. The test resulted in a head loss value that allowed the licensee to demonstrate that ECCS and CSS performance would not be adversely affected by the debris that might be generated and transported during a LOCA as long as that debris amount is bounded by that in the test. The strainer test provides the acceptance criteria for the amount of fine fiber that can arrive at the strainer and still have the strainer perform its design function acceptably. Any break or initiating event that generates and transports fine fiber in excess of the acceptable amount (191.78 lbm) is considered to fail and contribute to plant risk. Breaks that generate and transport less than the limit are assumed to result in acceptable ECCS and CSS strainer performance.

Although the test was performed using acceptable methods, the amount of fibrous debris included in the test was much less than what could be generated and transported during a worst-case LOCA. The licensee had assumed a 7D ZOI for fibrous debris generation based on testing that was later found to be non-conservative by the NRC staff. The licensee then reverted its fibrous debris ZOI to the value approved by the NRC staff (17D). Use of the larger ZOI would mean that some very large breaks could yield fibrous debris amounts not bounded by the 2008 test. The risk of having some breaks that may result in fibrous debris amounts greater than those tested is evaluated in the risk portion of this SE section 3.4.2.10, "Systematic Risk Assessment." This section evaluates the acceptability of the test methodology and the evaluation of the test results at the amounts of debris included in the test.

The licensee provided the head loss test results in a December 11, 2008, supplemental response for GL 2004-02.[75] Much of this information was also included in the licensee's August 20, 2015, RoverD submittal in Attachment 1-2.[20] The staff reviewed the supplemental responses to GL 2004-02 and the risk-informed license amendment request (LAR) submitted by the licensee.

The licensee's evaluation of head loss and vortexing was performed in accordance with NEI 04-07 guidance that includes the NRC staff SE,[48] and the revised NRC staff's review guidance for GL 2004-02 responses.[52] The licensee provided information requested in the Content Guide.

In order to determine the minimum strainer submergence, the licensee calculated the minimum water level. The minimum strainer submergence is 0.5 inches and 10 inches for SBLOCA and LBLOCA, respectively. The NRC staff evaluation of the water level calculation is performed in the NPSH section (SE Section 3.4.2.8.5).

The licensee provided the assumptions and results of its head loss evaluation. Assumptions include:

- The sump fluid is assumed to be saturated at the surface of the pool and no credit is given for subcooling. (Note that this is true for NPSH determination, but the licensee credits some accident pressure to suppress flashing across the strainer and deaeration of the fluid. This issue is the subject of questions 33 and 34.)

- Head loss is linearly proportional to dynamic viscosity.

- The strainer test inputs are scaled using a scaling factor defined as ratio of the surface area of the scale (test) strainer to the surface area of the full-sized strainer.

- The measured head loss occurred with relatively cold water; therefore, it was multiplied by a ratio of water viscosities between the cold and warmer water to obtain the predicted head loss at post-LOCA water temperatures.

These assumptions are consistent with the NRC guidance.

The licensee stated that the results of the evaluation show neither vortexing, air ingestion, nor significant voiding will occur. The licensee performed testing under conditions more conservative than those in the plant to determine whether the strainer was subject to vortex formation at limiting conditions. The testing determined that vortex formation is not a concern for the plant strainer. Some containment pressure was credited by the licensee to prevent flashing. Questions 33 and 34 follow up on the potential for flashing and deaeration and are discussed below.

The licensee performed head loss testing at the Alden Research Laboratory in July 2008 using a strainer module identical to those installed in STP containment. Two tests were performed: one for clean strainer head loss (CSHL) and one for debris loaded head loss. The licensee stated that a thin-bed test did not need to be performed because the final debris bed was less than 1/8-inch thick. The test apparatus included a flume, pumps, instrumentation, piping, chemical mixing tanks, a heat exchanger, and a strainer module to create prototypic plant conditions. The test strainer had a surface area of 91.44 ft^2. The flow rate and debris quantities were scaled based on the strainer surface area. The test flow rate was 353 gpm and the strainer approach velocity was 0.0086 ft/sec.

The licensee did not use reflective metal insulation in the July 2008 tests since previous testing showed stainless steel reflective metal insulation did not transport to the strainer in flow conditions prototypical of the plant. Miscellaneous debris such as lead blanket pieces and tags and labels were not included in the July 2008 testing for the same reason as reflective metal insulation. For fibrous debris, in the test, the licensee used NUKON™ as a surrogate for NUKON™, Knauf Elevated Temperature as a surrogate for Thermal-Wrap™, and NUKON™ as a surrogate for latent fibers. Surrogate materials were used for coatings. Acrylic powder was used for epoxy particulates and acrylic chips for epoxy chips. Tin powder was used as a surrogate for coatings. In SE Section 3.4.2.6.5, "Coatings," the NRC staff evaluated the licensee's choice of surrogates for coatings and found them acceptable. Other particulates the licensee included in the testing were Marinite® and Microtherm® powder. It should be noted that all Marinite® insulation was removed from the STP containments in 2009. The licensee also used a dirt/dust mixture made of silica sand for latent particulate debris. The amount of chemical debris used in the testing was based on a 30-day spray duration and the chemical surrogates were generated in mixing tanks. Chemicals used during the testing were Aluminum Oxyhydroxide (AlOOH) and Calcium Phosphate ($Ca_3(PO_4)_2$). Sodium Aluminum Silicate ($NaAlSi_3O_8$) was predicted to form inside containment following a LOCA, but the testing used AlOOH in lieu of $NaAlSi_3O_8$ due to its hazardous nature. The licensee's treatment of chemicals is evaluated in SE Section 3.4.2.8.6, "Chemical Effects," and found to be acceptable.

The debris amounts used in the July 2008 tests were scaled to represent the debris amount predicted to arrive at the strainers assuming that two trains (40 modules total) are operating. The type of debris, amount of debris, and test surrogate used are included in Attachment 1, pages 40-41, of the August 20, 2015, submittal. [20]

When the tests were performed, the licensee included the maximum amounts of debris calculated to arrive at the strainers based on material in containment at that time. After the testing was performed, all Marinite® was removed from containment and more Microtherm® was found than was included in the July 2008 tests. The licensee performed a defense-in-depth evaluation of the impact of decreasing Marinite® and increasing Microtherm® on head loss. The NRC staff did not find the defense-in-depth evaluation adequate and asked additional questions on this topic in SSIB-3-4 issued in RAI Round 3 on April 11, 2016. [67] The licensee provided additional information in response to this question, which is discussed below.

The licensee provided information regarding the methodology used to perform the test including debris preparation and addition, and the steps taken during testing. The licensee also provided photographs of some steps during the test, and of the strainer with debris on it after the test. The CSHL test took pressure drop readings for flow rates ranging from 176 gpm to 530 gpm. The CSHL from the test is 0.0923 ft at a flow of 353 gpm and temperature of 116.3 degrees Fahrenheit (°F). This CSHL value is used only for the test and is not the CSHL value of the strainer in the plant. The total CSHL for the plant strainer is determined by including head losses from several modules, other losses inherent in the strainer design, and losses from piping that was added as part of the strainer modification that connects the strainer to the pump suction. The calculated CSHL used the Performance Contracting Inc. (PCI), correlation, which is derived from test data. The corrected CSHL for the plant strainer was calculated as 1.95 ft at 128 °F.

The licensee stated that the head loss due to debris blockage was determined by measuring the differential pressure across the strainer and debris bed in the test facility. The head loss calculation included subtracting velocity head from the downstream pressure taps and CSHL from the pressure drop between these taps and the flume water surface. The maximum corrected head loss for the debris bed was 8.745 ft at a flow rate of 356 gpm and temperature of 116.3 °F. The head loss was decreasing with respect to time at the conclusion of the test, so the maximum head loss measured during the test was considered bounding. Therefore, the licensee determined that the result did not need to be extrapolated for the extended time that the strainer may have to remain in service following a LOCA. The fiber bed was less than 1/8-inch thick, therefore, the licensee did not perform a thin-bed test. Flow sweeps were performed to ensure bore holes did not form during testing. No vortices were observed during testing. The licensee credited near-field settling during testing. Computational fluid dynamics was used to model the debris transport and flow patterns near the strainer in a test flume configuration. This model was compared to a similar model of the plant in the vicinity of the strainers.

The licensee calculated total strainer head loss (TSHL) by first temperature correcting head loss values for CSHL and the debris head loss test results, then adding both values together. The flow velocity is 0.009 ft/sec (100 percent viscous flow through the debris bed). The TSHL is 3.8 ft at a post-LOCA temperature of 267 °F.

The licensee determined that it could use the amount of fine fibrous debris included in the head loss test as an analytical limit to compare to debris generation scenarios (LOCAs) that may occur at STP.

The NRC staff reviewed the licensee's evaluation and found that the testing and evaluation of test results were conducted generally in accordance with NRC staff-accepted practices. However, in some areas, the NRC staff needed additional information to ensure that the head loss and vortexing area was evaluated adequately. The NRC staff notes that at the time the test was performed, testing guidance had not been fully developed by the staff or by industry. Because it was relatively early in the development of test practices, the NRC staff frequently asked licensees clarifying questions about head loss test methods, assumptions, and analyses. After the NRC staff received clarifying information from the licensee, it determined that the licensee performed the head loss evaluation consistent with NRC staff guidance. More details on the staff evaluation are provided below.

The licensee's claim that a thin-bed test did not need to be performed because the final bed thickness was less than 1/8-inch thick was not initially acceptable to the NRC staff. NRC guidance states that thin beds are to be fibrous beds fully saturated, to the extent possible based on plant-specific debris amounts, with particulate debris. The 1/8-inch value referenced by the licensee for thin-bed determination was based on uncompressed, or 'as manufactured,' fiber density. More recently the 1/8-inch value has been rejected as a thin-bed metric. Fiber in a debris bed is compressed such that 1/8-inch of uncompressed fiber would be much thinner once deposited on a strainer with particulate debris. However, the NRC staff reviewed the test data and determined that once chemical debris was added to the test, that the debris bed was saturated by particulates, and it was unlikely that a higher head loss would occur with a bed containing less fiber. Thus, the NRC staff determined that the licensee did not need to perform additional thin-bed testing.

The licensee's test credited near-field settling which means that not all of the fiber added to the test was transported to the strainer during the test. A significant portion of the fiber settled in the flume upstream of the strainer. The NRC staff position is that test methods should attempt to transport all debris in the test to the strainer. This ensures that the test condition is consistent with the condition predicted by the transport evaluation. The NRC staff allows near-field settling in testing if the licensee can demonstrate that the velocities and turbulence in the testing are realistic or conservative with respect to the plant. The NRC staff asked several questions regarding the allowance for near-field settling and found that the test conditions appropriately accounted for the issues in question. These issues are discussed in SE Section 3.4.2.7, "Debris Transport," because they are associated with transport in the plant and transport in the test.

Although the test was performed assuming that two trains of ECCS were operating, the licensee assumed that the limiting load for evaluation of a single-train case is one-half of the amount included in the test. The NRC staff determined that this is a reasonable assumption because the acceptance criterion is based on fine fiber. Therefore, since a single train has one-half of the area of two trains and flow conditions are identical between trains, a single train could take one-half of the fiber load of the two-train case. The NRC staff also considered the particulate and chemical debris loading on the strainers and determined that because the head loss during the test did not increase with additional loads of chemicals at the end of the test, it was unlikely that additional particulate in the single-train debris bed would have a significant effect on head loss. Therefore, the NRC staff finds the licensee's assumption that one-half of the fiber load for the two-train case is a valid acceptance criteria for the single-train case.

The NRC staff used verifications performed by SwRI to validate that the licensee's inputs for SE Section 3.4.2.8.3, "Head Loss and Vortexing, were determined accurately. The contractor used a combination of confirmatory calculations, engineering review, and exercising of the licensee's software to perform the verifications. The results of SwRI's review allow the NRC staff to conclude, with a high level of confidence, that a significant portion of the inputs to the head loss testing were conducted properly because SwRI confirmed that the licensee's inputs were accurate.

On December 23, 2009,[66] the NRC issued an RAI to the licensee regarding head loss and vortexing. On August 20, 2015,[20] the licensee responded to the RAI. In question 27, the NRC staff asked the licensee to provide the vortex test conditions and observations and describe a discrepancy in the reported Froude (Fr) number used to estimate the potential for vortex formation at the strainer. The NRC staff reviewed the response and found it adequate because the licensee provided a reference to testing that shows that vortex formation will not occur under the conditions in which the strainers are required to operate at STP.

In question 28, the NRC staff asked the licensee to provide the debris sizing, amount of each debris type included for each size category, and basis for the size distribution chosen for debris surrogates added to the head loss testing. The licensee used surrogate materials that have been accepted by the NRC staff for strainer head loss tests. The licensee provided the sizing for the surrogates used in the testing as requested, therefore, the response is adequate. The NRC staff evaluation of coatings materials used in the testing is included in SE Section 3.4.2.6.5, "Coatings." This issue is also discussed in the SE Section 3.4.2.6.3, "Debris Characteristics."

In question 29, the NRC staff requested that the licensee justify the debris addition sequence and confirm that it did not affect the ability of more transportable debris to reach the strainer in a non-conservative fashion. The licensee provided information demonstrating that the surrogates were added to the test from most transportable to least transportable and that transport of the debris surrogates to the strainer was not inhibited due to addition order. This meets NRC guidance. Therefore, the NRC staff finds the response acceptable.

In question 30, the NRC staff asked the licensee to provide the head loss plots for the testing including annotation of significant events during the test. The licensee provided a plot of head loss versus time for the STP design basis test. The NRC staff reviewed the response and determined that it provided the information requested.

In question 31, the NRC staff asked the licensee to answer the questions posed in (a) through (f) below:

(a) Provide the design maximum head loss and its basis.

The licensee performed a single head loss test and determined that the head loss from the test was 9.35 ft. This test was conducted at a relatively low temperature. The licensee corrected the head loss to design basis conditions when the sump fluid will be significantly hotter. The head loss for the design basis condition is 5.71 ft based on a correction due to viscosity changes between the temperatures. The NRC staff finds that these results are reasonable based on the head loss testing and correction for the viscosity of water.

(b) Verify the structural pressure limit of 5.71 ft is not exceeded during LOCA response.

The licensee demonstrated that the pressure drop may exceed 5.71 ft. when the sump cools. However, the licensee also demonstrated that the structural integrity of the strainer is not challenged at lower temperatures due to higher material strengths at lower temperatures. The NRC staff concludes that the response is adequate since the strainer structural integrity is not challenged at higher pressures due to the increased material strength at lower temperatures.

(c) Provide head loss at lowest postulated sump temperature and compare it to structural limit.

The licensee provided a maximum head loss of 9.35 ft at 128 °F. The NRC staff confirmed that the licensee used the correct head loss input for performing the structural evaluation. Since the licensee demonstrated that the structural integrity of the strainer is not challenged at lower temperatures due to higher material strengths at lower temperatures, then the head loss is within the structural limit. Therefore, the response to the question is acceptable.

(d) State whether CSHL counts against structural limit, or if only debris head loss needs to be considered.

The licensee included CSHL in the structural limit. The NRC staff reviewed this response and finds it acceptable because it includes CSHL which increases the differential pressure (and therefore, structural loading) in the analysis and is conservative.

(e) Clarify whether TSHL of 6.504 ft at 171 °F includes CSHL.

The licensee confirmed in Attachment 1 of its August 20, 2015, submittal in Item 3.g.16 that the TSHL of 6.5 ft at 171 °F includes CSHL.[20] This value was confirmed by an independent calculation by the NRC staff. The response is acceptable since the licensee confirmed that CSHL was included in the TSHL.

(f) Provide the results of extrapolations of the head loss test results to various temperatures required for head loss considerations.

The licensee provided a graphical representation of this information. The NRC staff finds this response is acceptable.

In question 32, the NRC staff asked the licensee to provide information on whether the strainer is vented. The licensee response stated that the emergency sump strainers are not vented. The NRC staff reviewed the response and finds it acceptable and that the strainer evaluations used assumptions consistent with an unvented strainer.

In question 35, the NRC staff asked the licensee to address the potential for floating debris to collect on top of the strainer during a SBLOCA and thus provide a potential air-entrainment pathway to the interior of the strainer. The NRC staff reviewed the response and finds it acceptable because the licensee demonstrated that there is not any debris type postulated in containment likely to float for extended periods following a LOCA.

In question 36, the NRC staff asked the licensee to provide a technical basis for concluding that drainage of spray water near the strainer surface will not result in splashing and surface disturbances that would cause unacceptable air entrainment into the strainers and ECCS containment spray sumps considering the minimal strainer submergence for SBLOCAs. In the response, the licensee clarified that runoff paths do not extend all the way to the containment wall, but rather stop at the structural support column. Based on the clarification of the location of the runoff and the presence of a structure between the runoff and strainer, the NRC staff concludes that air entrainment due to runoff in the vicinity of the strainer is not a concern for the STP strainer. This is also discussed in SE Section 3.4.2.7, "Debris Transport," therefore, the NRC staff's concern in question 36 is addressed.

In question 33, the NRC staff asked the licensee to provide the margin to flashing and the assumptions for the calculation. The licensee provided a calculation for the margin to flashing for LBLOCAs at the start of recirculation as containment pressure + submergence – TSHL – vapor pressure = 43.1+0.3-1.5-39 = 2.3 psi. The licensee stated that post-LOCA containment-pressure credit is needed to eliminate the potential for flashing. The licensee stated that because the minimum strainer submergence was conservatively determined to be 0.5-inch for SBLOCAs, sump temperature and containment pressure would be lower for a SBLOCA than LBLOCA, strainer flow rate would also be lower, and debris transported to strainers would be much less such that there would be open strainer areas. Therefore, the licensee concluded that flashing is not expected to be an issue for SBLOCAs.

The NRC staff could not determine the margin that existed in the licensee's calculations for containment pressure and sump temperature. The NRC staff guidance states that when containment pressure is credited to suppress flashing, the pressure should be calculated conservatively low and sump temperature should be calculated conservatively high, or a small portion of containment pressure may be applied to ensure significant margin exists. In addition, the NRC staff was unclear what head loss was assumed in the SBLOCA case. Because submergence is very low, a small head loss may have an impact on the flashing evaluation. It was not clear that the reduced submergence and potential for some head loss was taken into account for the SBLOCA case. In Attachment 1-4 of its letter dated October 20, 2016,[30] the licensee provided additional information regarding these issues. The licensee clarified that the sump temperatures were calculated using design basis calculations that bias the temperature high. The licensee provided additional cases that show that the cases with the high sump temperature, and containment pressure biased low, result in adequate margin to flashing. The specific cases were provided in response to question 34 and are discussed in more detail below. Based on the above, the NRC staff concludes that flashing will not occur at the debris loads evaluated.

In question 34, the NRC staff asked the licensee to provide an evaluation of the potential for deaeration of the fluid as it passes through the debris bed and strainer and whether any entrained gasses could reach the pump suction. The licensee stated that the net void fraction is zero percent, and therefore, void fraction is not an issue for any of the pressure and temperature combinations associated with the post-LOCA fluid. The licensee explained that any void fraction that could occur at the strainer debris bed is minimal and that if any should occur, it is reversed before strainer discharge water leaves the sump due to the significant static head of water above the ECCS and CSS pump suction inlets within the sump. The licensee concluded that the net void fraction is therefore zero and not problematic for any conditions.

The NRC staff did not understand the licensee's assertion that voiding would be reversed by the time the fluid reached the pump suction. The staff agrees that this would be true for flashing, but staff guidance states that flashing should not be allowed to occur. For deaeration, the NRC staff was unclear how the process would be reversed. The staff also did not understand the margin included in the calculations for deaeration, or whether the reduced submergence and head losses for the SBLOCA case was accounted for, similar to the issue discussed above in question 33.

In its October 20, 2016, submittal,[30] the licensee provided additional information regarding these concerns. The licensee revised its analysis to remove the assumption of voiding reversal. The licensee explained that the evaluation was performed using the minimum containment pressure available to prevent the void fraction from exceeding 2 percent (the limit in staff guidance). The licensee provided updated cases for sump temperature and containment pressure. The licensee evaluated the temperature and pressure at the earliest possible time of recirculation, but stated that actual switchover times are estimated to be about double the shortest time. The longer time to the initiation of recirculation would result in significant subcooling of the fluid before the strainers are placed in service and would eliminate concerns with flashing and reduce deaeration. The licensee provided cases that calculate deaeration using different assumptions for sump temperature and pressure. All cases showed margin to flashing, and only two of the cases resulted in deaeration greater than the amount in staff guidance. These two cases were conducted only to show the point at which deaeration would become excessive. The cases were determined to be unrealistic and additional analysis was provided to show that deaeration would be maintained at an acceptable level. The licensee provided calculations that show, even with deaeration, that NPSH margins are maintained for all cases within the head losses determined via testing. Therefore, the NRC staff concludes that the issue of deaeration has been adequately addressed and the response is acceptable.

During its review of the RoverD evaluation, the NRC staff could not determine that the amounts of particulate debris included in the 2008 test were bounding of all breaks that could occur at STP. In response to question SSIB-3-4 regarding how the particulate debris amounts in the test bounded the amounts that could be generated in the test, the licensee provided additional information. The following table is a summary of the particulate debris amounts calculated to transport to the strainer from the most limiting break, the amounts included in the test, and the margin. Note that all Marinite® was removed from within the containment buildings by the licensee following the 2008 testing.

Particulate Debris	Rover D Maximum Transported Volume (ft^3)	Tested Volume (ft^3)	Margin (ft^3)
Microtherm®	0.959	2.200	1.241
Marinite®	0	3.972	3.972
Coatings and Latent Particulate	16.591	11.362	-5.229

The negative margin shown for coatings and latent particulate is compensated for by the Marinite® and excess Microtherm® that was included in the test. This table shows that the 2008 test debris load bounded the largest amount of particulate that is predicted to be generated and transported from any LOCA break within the STP containments. The NRC staff concludes that it is acceptable to substitute Marinite® as a surrogate for coatings debris on a mass basis. The staff conclusion is based on two factors. First, Marinite® is known to be a problematic debris type, very similar to Cal-Sil. Second, Marinite® is less dense than coatings debris so that it takes up more volume in the debris bed resulting in the potential for higher head losses. The NRC staff concludes that the amounts of particulate debris included in the 2008 test were bounding of all potential particulate debris loads that could be generated and transported within the STP containments and the response to this question is acceptable.

In SSIB-3-5, the NRC staff requested that the licensee justify its use of only fine fiber as the criterion for whether initiating events would result in acceptable strainer performance. The licensee stated that testing determined that fine fibers are the only type that will transport and collect on the strainer surface. The licensee stated that the analytical debris generation evaluation found that most small fiber and all large fiber pieces would settle in the sump pool away from the strainer. Although the transport evaluation found that some small fiber pieces would transport to the strainer, testing in a conservative representation of STP flow conditions demonstrated that the small pieces would not actually transport to the strainer surface. The staff considers the flow conditions in the test to be conservative compared to those in the plant. That is, the test condition was more conducive to debris transport than the plant condition. The NRC staff finds the response acceptable. Issues related to this question and the potential for the erosion of small fibers are discussed in the SE Section 3.4.2.7, "Debris Transport."

In SSIB-3-6, the NRC staff requested that the licensee clarify whether the coatings within the reactor cavity were included in the head loss testing for breaks that could occur within the reactor cavity. The licensee stated that the coatings are considered degraded qualified coatings. The licensee identified that for breaks within the reactor cavity, the coatings are treated as particulate debris, and for breaks outside the reactor cavity the coatings are treated as degraded qualified coatings that fail as chips and particulate. The failure mode of the coatings is discussed in SE Section 3.4.2.6.5, "Coatings." For SE Section 3.4.2.8.3, "Head Loss and Vortexing," the staff determined that the testing accounted for the appropriate amount of coatings. This is related to SSIB-3-4, which is discussed above. The NRC staff concludes that the coatings were represented appropriately in the 2008 test and, thus, the question is addressed.

The STP evaluation calculated a fine fibrous debris amount arriving at the strainer for each of the potential initiating events that could reasonably lead to sump recirculation. The methodology for calculating the amount of debris predicted to arrive at the strainer is discussed in SE Sections 3.4.2.6, "Debris Source Term" and 3.4.2.7, "Debris Transport." The 2008 head loss test provides the acceptance criterion for the amount of fiber that can arrive at the strainer before it is considered to fail. Two cases were considered. One case for two ECCS trains running and one case for a single ECCS train running. The single case results in more failures because the strainer area is one-half of the two-train case so only one-half the amount of debris is required to block the strainer. The NRC staff concludes that the use of the STP 2008 test for determination of the fine fiber debris acceptance criterion is acceptable for all potential pump operating combinations. The NRC staff also concludes that the use of only fine fiber as the acceptance criterion, not other debris types or sizes, was appropriate based on the test and evaluation methodology used by STP.

The NRC staff concludes that the maximum tested debris head loss represents a value that is bounding for the STP strainer for the debris load tested. This conclusion is based on the test being performed using conservative inputs and test methods that have been accepted by the NRC staff. The staff also finds that the measured head loss does not need to be extrapolated to future times because the head loss in the test was decreasing at test termination and the licensee used the maximum head loss measured during the test.

The NRC staff reviewed the licensee's evaluation against the staff-accepted guidance and concludes that the licensee has appropriately determined the head loss across the sump strainer for the debris load tested. The licensee has shown that the potential for formation of a vortex at the strainer does not exist under the plant-specific conditions at STP. The licensee has demonstrated that the strainer will perform acceptably under postulated LOCA conditions, limited by the amount of debris represented in the 2008 test. Therefore, the NRC staff concludes that the licensee's evaluation of head loss and vortexing is acceptable.

3.4.2.8.4 Sump Structural Analysis

In Enclosure 4-3, Section 2.2.25, "Strainer Structural Margin," of the licensee's submittal dated November 13, 2013,[4] the licensee stated that the strainers have been structurally qualified for head losses up to 4.00 psi differential pressure at 128 °F, which is equivalent to a head loss of 9.35 ft. In Enclosure 3, Appendix 6A of the November 13, 2013, submittal, the licensee noted that the structural analysis was previously completed in earlier GL 2004-02 sump performance evaluation activities and was documented in correspondence to the NRC dated December 11, 2008.[75] In the 2008 letter, the licensee provided information requested in the NRC's Revised Content Guide for GL 2004-02 Supplemental Responses.[51]

Based on the information provided in the 2008 letter, the NRC staff issued several rounds of RAIs which culminated in the licensee's August 20, 2015,[20] supplement. This supplement incorporated the previous RAI responses, as well as the original information provided in the 2008 response. Unless otherwise noted, the following staff evaluation addresses the information provided in the August 20, 2015, LAR supplement.

The licensee stated that the new strainers were designed for loads due to weight, pressure, and dynamic loads due to seismic and sloshing. The pressure load is the differential pressure across the strainers' perforated plates in the operating condition and two cases were analyzed; Case 1 is at the start of recirculation with a low differential pressure and a high temperature, while Case 2 is 30 days post-accident with a higher differential pressure and a lower temperature. The pressure load on the strainer was identified as 5.71 ft. of head for Case 1 and 9.35 ft. of head for Case 2. The licensee noted that the debris loading was the same for both cases and that the changes in head loss were due to temperature effects. Case 2 was derived from plant-specific testing, while Case 1 was an estimate which was shown to be conservative by the testing. Thermal expansion loads were taken as zero because the strainers are free to expand without restraint. The licensee further stated that the load combinations were based on the STP design requirements.

The strainers were designed in accordance with the American Institute of Steel Construction (AISC).[76] The acceptance criteria also came from the AISC 7th Edition, when applicable. American National Standards Institute/AISC N690-1994,[77] was used to supplement AISC 7th Edition for areas related specifically to stainless steel. For the perforated plates, the licensee noted that the equations from Appendix A, Article A-8000 of the ASME Code, Section III, 1998 Edition, were used because the equations are written specifically for perforated plates.

The licensee provided a table summarizing the actual and allowable stresses along with the interaction ratio for components of the sump assembly. The interaction ratio is the actual load divided by the allowable load, and an interaction ratio value less than one demonstrates the component meets the stress requirements of the applicable code. The licensee provided interaction ratio values for both of the load cases discussed above, and all of the values were less than one.

However, the original risk-informed submittal dated June 19, 2013,[1] noted that strainer structural failure was a possible failure mode analyzed in the probabilistic risk analysis, indicating that there were load conditions where the strainer could fail and the interaction ratio values would be greater than one. To address this apparent discrepancy, the NRC staff issued an RAI on April 15, 2014,[78] requesting additional information on the conditions that could fail the strainers. In its response dated July 15, 2014,[17] the licensee stated that the results represented a deterministic analysis demonstrating that the strainers do not fail under any design basis loading condition. The licensee further explained that the probabilistic risk analysis considers that failure is always possible and attempts to quantify that probability. For the probabilistic risk analysis, a structural "failure" was identified as any scenario with a differential pressure greater than 9.35 ft., which was the highest analyzed pressure in the structural qualification calculation.

With regard to potential loadings associated with a HELB, the licensee stated that there are no high-energy lines in the area of the emergency sumps except for the HHSI lines which are used for accident mitigation and are not assumed to be the accident initiator. For this reason, no evaluations were needed for HELB.

The licensee stated that the new strainer design does not involve backflushing.

The NRC staff reviewed the licensee's information, and determined that the use of the identified codes, and the associated load combinations, is acceptable because they are included in STP's UFSAR and listed in SRP Section 3.8.3 as acceptable design codes for containment internal structures.[53]

The NRC staff also reviewed the information provided for both loading cases and finds reasonable assurance that the two load cases envelope all possible loads on the strainer. Based on the plant-specific test results, Case 1 is a conservative estimate of the differential pressure on the strainer. Case 2 uses the same maximum debris loading as Case 1, with the only pressure difference being due to temperature effects. As the water temperature decreases and increases the differential pressure, the yield stress of the steel also increases. Since both cases assume the maximum debris loading, and both cases are acceptable (interaction ratio less than 1.0, see evaluation of structural qualification and margin discussed above), there is reasonable assurance that the two analyzed conditions bound all possible debris loads and temperature combinations.

The NRC staff reviewed the RAI response and noted that according to a deterministic analysis using acceptable code allowable values, the strainer will not fail under any design basis loads. The clarification provided by the licensee, along with the tables demonstrating interaction ratio values less than 1.0, provide the NRC staff with reasonable assurance that the sump strainer assemblies will remain structurally adequate under loading conditions that apply a differential pressure less than or equivalent to a head loss of 9.35 ft., which is the maximum design basis load.

The NRC staff reviewed the licensee's December 11, 2008, submittal and verified that the only high-energy lines in the area of the emergency sump are the HHSI lines. The NRC staff finds that a HELB evaluation is not necessary for these lines because they are not pressurized during normal operation and are not assumed to be the accident initiator.

The NRC staff notes that backflushing is not credited in the new strainer design, and therefore no structural analysis was necessary to address reverse flow.

Based on the NRC staff's review of the licensee's submittal of December 8, 2011, and the risk-informed LAR of June 19, 2013, along with supplemental information provided in response to NRC staff RAIs, the NRC staff finds that the licensee has successfully provided the information requested for sump structural analysis. Therefore, the NRC staff concludes that the sump strainer is structurally acceptable for the assumed design basis loads associated with the maximum equivalent head loss for which it is deterministically qualified (9.35 ft). Loading of the strainer beyond its structural qualification is assumed to result in a failure of the sump strainer. The risk associated with a failure of the strainer is evaluated in SE Section 3.4.2.10, "Systematic Risk Assessment."

3.4.2.8.5 Net Positive Suction Head

The licensee stated the pump flow rates for each operating train are 2,800 gpm for LHSI, 1,620 gpm for HHSI, and 2,600 gpm for CSS. The sump flow rate for a LBLOCA is 7,020 gpm per strainer. Sump temperature for this license amendment was evaluated at 267 °F and 128 °F, which represent maximum and 30-day-post-LOCA sump temperatures. In its RoverD supplement,[20] the licensee calculated the sump temperatures in the containment analysis using GOTHIC [Generation of Thermal-Hydraulic Information for Containments], which yielded values of 275 °F and 125 °F for the maximum and 30-day-post-LOCA cases. The licensee stated the effects of these differences in temperature were considered in the evaluation for this LAR and were found to have little or no impact. The minimum containment water level at the start of recirculation is 38 inches above the floor for an LBLOCA.

The licensee used the following assumptions when calculating sump flow rate, pump flow rates, sump temperature, and minimum containment water level:

- Safety injection pump flow rates are the maximum values given in the technical specification

- CSS pump flow rate is based on calculated maximum values with two trains operating

- Sump temperature from the containment analysis maximizes the sump temperature by using the maximum temperatures for cooling water to the heat exchangers and for the water of the ultimate heat sink

- Containment water level used conservative input values for pool contributions and accounted for holdup in the containment, filling of empty piping, water in transit, and steam holdup.

The licensee stated that the basis for the required NPSH values came from the test curves supplied by the pump vendor. At the centerline of the pump suction nozzles, the required NPSH is 1.5 ft. for LHSI, 1.1 ft. for HHSI, and 1.4 ft. for CSS.

The licensee described the LBLOCA system response scenarios. The safety injection and CSS pumps start automatically and take suction from the refueling water storage tank. When the tank is drawn down, the pumps' suction is automatically switched over to the containment sumps for the recirculation mode. The HHSI and CSS pumps may be turned off later in the post-accident mitigation per the emergency operating procedures.

The licensee stated that the system response for an SBLOCA scenario is similar to that for the LBLOCA. However, the LHSI pumps may not be able to inject if the RCS pressure is high. In that case, the HHSI pumps are used for RCS inventory control during the injection and the recirculation phases. The CSS pumps are assumed to be in operation due to automatic actuation if their containment pressure set point is reached.

The evaluation assumes that all ECCS and CSS pumps are available at recirculation. After recirculation is initiated, two of the three trains of CSS are assumed to be operating as one is secured per the licensee's emergency operating procedures. The licensee stated that if there is a single failure of one ECCS and CSS train, two operating trains (and two strainers) will remain available.

The minimum containment water level was calculated using a correlation developed by the licensee. The correlation accounts for the volume of the lower containment including water displaced by solid objects. The correlation also accounts for water sources and water that is held up and may not reach the containment pool. The SBLOCA case results in a lower water level because it is assumed that the RCS is refilled with water and the accumulators do not discharge.

Although some containment accident pressure was credited in the head loss and vortexing analysis to prevent flashing, the licensee does not credit containment accident pressure in STP's analysis of NPSH. For a discussion of the containment accident pressure credited for suppression of flashing see the SE Section 3.4.2.8.3, "Head Loss and Vortexing."

The NPSH margin was reported by the licensee for varying sump temperatures. See the tables below. The NPSH available values do not include strainer head loss. Therefore, the actual margin is calculated by subtracting the TSHL from the NPSH margin. The TSHL given below includes the debris bed head loss and the CSHL. For sump temperatures above 212 °F, the NPSH available considered that the containment pressure was equal to the vapor pressure. For sump temperatures below 212 °F, the containment pressure was taken as 14.7 psi absolute (psia).

Low Head Safety Injection Pump				
Sump Temperature, °F	NPSH Required, ft.	NPSH Available, ft.	NPSH Margin, ft.	Total Strainer Head Loss, ft.
267, Start of Recirculation 24 minutes	1.5	7.5	6	3.8
226, Hot Leg Switchover 5.5 hours	1.5	7.5	6	4.6
215	1.5	7.5	6	5
212	1.5	7.5	6	5.1
210	1.5	8.7	7.2	5.1
206	1.5	11.1	9.6	5.2
200	1.5	15.4	13.9	5.4
190	1.5	20.1	18.6	5.8
171, 24 hours	1.5	27.9	26.4	6.5
128, 30 days	1.5	37	35.5	9.2

High Head Safety Injection Pump				
Sump Temperature, °F	NPSH Required, ft.	NPSH Available, ft.	NPSH Margin, ft.	Total Strainer Head Loss, ft.
267 Start of Recirculation 24 minutes	1.1	7.4	6.3	3.8
226 Hot Leg Switchover 5.5 hours	1.1	7.4	6.3	4.6
215	1.1	7.4	6.3	5.0
212	1.1	7.4	6.3	5.1
210	1.1	8.8	7.7	5.1
206	1.1	11.2	10.1	5.2
200	1.1	15.5	14.4	5.4
190	1.1	20.2	19.1	5.8
171 24 hours	1.1	28.0	26.9	6.5
128 30 days	1.1	37.2	36.1	9.2

Containment Spray Pump				
Sump Temperature, °F	NPSH Required, ft.	NPSH Available, ft.	NPSH Margin, ft.	Total Strainer Head Loss, ft.
267 Start of Recirculation 24 minutes	1.4	7.2	5.8	3.8
226 Hot Leg Switchover 5.5 hours	1.4	7.2	5.8	4.6
215	1.4	7.2	5.8	5.0
212	1.4	7.2	5.8	5.1
210	1.4	8.4	7.0	5.1
206	1.4	10.8	9.4	5.2
200	1.4	15.1	13.7	5.4
190	1.4	19.8	18.4	5.8
171 24 hours	1.4	27.6	26.2	6.5
128 30 days	1.4	36.7	35.3	9.2

On December 23, 2009,[66] the NRC staff issued RAIs regarding NPSH. On August 20, 2015, the licensee responded to the RAIs.[20]

In question 39, the NRC staff asked the licensee to provide the volumes of the water sources that contribute to the formation of the containment pool for the limiting minimum containment water level, including specific discussions of both the large and SBLOCA cases. In response, the licensee provided volumes of water sources that contribute to formation of the containment pool for limiting minimum containment water level. The licensee also discussed SBLOCA cases and provided a table showing RCS holdup for LBLOCA, surge line LOCA, and SBLOCA. The NRC staff reviewed the response and finds it acceptable as it provided the requested information.

In question 40, the NRC staff asked the licensee to identify the methodology and any computer codes used to perform the suction piping friction loss calculations to determine the loss coefficients. In the response, the licensee stated that the suction piping and fitting friction head losses are based on standard industry methodologies. The licensee also stated that the maximum pump flow rates were used and no computer codes were used for the calculations. These are industry standard methods that are acceptable to the NRC. The NRC staff reviewed the response and finds the licensee's use of standard methodologies is acceptable.

In question 41, the NRC staff asked the licensee to state the criterion and methodology used by the pump vendor to determine the net positive suction head required ($NPSH_R$) for all pumps taking suction from the ECCS sumps. The licensee stated that the $NPSH_R$ values for the pumps were determined by the vendor using testing. The NRC staff reviewed the response and finds it responsive.

In question 42, the NRC staff asked the licensee to provide the basis for considering the two-train NPSH results (based on the failure of one diesel generator) to be the limiting single failure. The NRC staff noted other cases, such as the operation of three trains (no single failure), or the operation of a single train (permitted by emergency operating procedures through operator actions to shut off redundant pumps). The RAI requested that the licensee provide the NPSH results for these other cases and the basis for considering the two-sump case as limiting with respect to NPSH margin. In the response, the licensee stated that its design basis is a minimum of two out of the three trains of ECCS and CSS be used for accident mitigation. For three-train operation, each of the three operating sump strainers will have a debris loading per sump less than the design case with two sumps operating. Consequently, the debris head loss will be less, which will have a positive effect on NPSH margin. The licensee stated that the use of a single train is not part of the deterministic design and licensing basis. If CSS pumps and/or HHSI pumps are secured, then there is less flow through the sump strainer, resulting in less debris head loss and less piping friction loss. This would have a positive effect on NPSH margin. The licensee concluded that the two operating sump case is the limiting case for NPSH margin. The NRC staff reviewed the response and finds it acceptable. The licensee considered the single-train case for the risk-informed evaluation. The single-train case considered that the hydraulic calculations would be the same as the two-train case, but a failure would occur with one-half of the debris load allowable for the design basis two train case. The NRC staff finds that these assumptions are acceptable.

In question 43, the NRC staff asked the licensee to state whether the NPSH results in its supplemental response dated December 11, 2008,[75] include debris bed and CSHLs. Updated NPSH results are provided in the tables above. The NRC staff concludes that these results show the TSHL, which may be subtracted from the NPSH margin to determine the margin including CSHL plus debris head loss.

In question 37, the NRC asked the licensee to provide NPSH margin results for LHSI, HHSI, and CSS pumps, for the LBLOCA and SBLOCA cases, under conditions of hot-leg recirculation. The licensee provided the NPSH margin for LBLOCA for LHSI, HHSI, and CSS pumps. The licensee claimed that its SBLOCA scenario would result in little to no debris on the strainer so that debris head loss would be very small. The licensee concluded that the only head loss for the SBLOCA case is CSHL. The licensee stated that the lower flow for the SBLOCA case would reduce CSHL compared to LBLOCA and the NPSH available would be slightly higher for a SBLOCA since piping friction loss is less due to lower flow. So, for the SBLOCA compared to the LBLOCA, NPSH margin would increase somewhat and TSHL would be much less. The licensee concluded that the SBLOCA case would be bounded by the LBLOCA case even though the pool level could be about 10 inches lower (resulting in smaller NPSH contributed by elevation), because the debris and CSHL would be significantly lower. The NRC staff determined that the SBLOCA case is bounded by the LBLOCA case except that the debris head loss would be zero for the SBLOCA case (see Attachment 1 of the licensee's letter dated June 16, 2016[24]). The licensee provided additional information in follow-up responses to question 37 in its October 20, 2016, supplement.[30] The licensee stated that the SBLOCA fibrous debris loads would be comprised almost entirely of latent debris and that the theoretical bed thickness on the strainer would be so small that a continuous debris bed would not cover the strainer. The NRC staff determined that the debris bed for the SBLOCA would not result in significant head loss and that the LBLOCA case bounds the SBLOCA case for NPSH because sump level only decreases by 9 inches. Thus, the NRC staff determines that this response is acceptable. However, the NRC staff asked additional questions regarding flashing and deaeration for the SBLOCA case. These issues are discussed in questions 33 and 34 in SE Section 3.4.2.8.3, "Head Loss and Vortexing."

In question 38, the NRC staff asked the licensee to describe the methodology and assumptions used to compute the limiting pump flow rates for all pumps taking suction from the ECCS sumps. The licensee stated that the CSS pumps discharge to a common ring header piping arrangement and the flow used for the NPSH evaluation is based on two CSS pumps operating resulting in higher flow per pump than if all three were operating. Flow rates used for LHSI and HHSI pumps are maximum values per technical specifications. The NRC staff accepted the justification for the LHSI and HHSI pump flow rates, but requested the licensee to provide more detail on how it calculated the CSS pumps' flow rates. In a follow-up response to question 38, the licensee stated that the CSS pumps' flow rates were determined using standard calculational methods using a hand calculation. The NRC staff accepted this response and concludes that the use of the two-train configuration is limiting for the calculation of NPSH margins for the CSS pumps. Because the staff accepts industry standard calculational methods in these cases, the response is acceptable.

The NRC staff notes that the CSS pump two train configuration is limiting from a design basis NPSH perspective, but the risk evaluation also requires consideration of the single train case. In response to questions from the Advisory Committee on Reactor Safeguards (ACRS) subcommittee, the licensee provided the NRC staff with an evaluation of the single CSS pump condition in a letter dated April 20, 2017.[79] The licensee stated that the CSS single pump flow rate would be 200 gpm higher for a total of 2800 gpm. This would result in a slight increase in strainer head loss and a slight increase in CSS pump NPSH required. These changes result in a negative NPSH margin if conservative design basis inputs and assumptions are maintained. Following the ACRS subcommittee meeting, the licensee noted that if realistic inputs are used for either chemical effects timing or sump level, then the NPSH margins would remain positive.

The licensee's April 20, 2017, letter also included a sensitivity study showing the effect on risk (ΔCDF and ΔLERF) of assuming core damage for all breaks 6 inches or larger in the single train configuration. This sensitivity showed that the effect of this assumption increased the calculated ΔCDF and ΔLERF by less than 1 percent, meaning that the risk acceptance guidelines were still met. The staff's contractor also performed a sensitivity study, which confirmed that the risk acceptance guidelines would be met even under the more conservative assumption that core damage would occur for all breaks 2 inches or larger in the single train configuration.

Based on the information provided by STP, and the sensitivity study performed by the staff's contractor, the staff concludes that the overall risk calculated by the licensee is not impacted.

In question 44, the NRC asked the licensee to identify the volume of holdup assumed for the refueling canal and provide further information that justifies that the refueling canal drains cannot become fully or partially blocked such that additional hold up could occur, or to provide the extent to which hold up could occur. The licensee provided a detailed explanation for why its refueling cavity drain lines will not become blocked. The licensee concluded that the refueling cavity drain lines are not assumed to become blocked and there is no water inventory holdup assumed other than the water below the elevation of drain lines. The NRC staff reviewed the response and was unable to determine the basis for the claim that large pieces of debris could not block the drains. The NRC staff asked the licensee to provide the basis for the claim that large pieces will not reach the refueling canal. The licensee responded in a follow-up to question 44. In response, the licensee stated that the grating in containment would prevent any large pieces of debris from reaching the refueling canal. The licensee provided CAD drawings to show the locations of the gratings. The NRC staff reviewed the response and finds it acceptable because the containment gratings will prevent debris from reaching and blocking the refueling cavity drain lines.

In SSIB-3-7, the NRC staff asked the licensee to clarify the design basis for the containment pressure and temperature. The licensee's responses, included in the NPSH section of the RoverD evaluation, stated that the GOTHIC computer code was used, but that the STP UFSAR currently references CONTEMPT4/MOD5 as the code of record. The licensee stated that the code of record was being changed from CONTEMPT4/MOD5 to GOTHIC. The licensee further stated that the evaluation using the GOTHIC code resulted in more limiting sump temperatures and did not significantly affect sump level. The licensee also stated that the effect of the temperature change on NPSH calculations was insignificant. The NRC staff reviewed the response and finds it acceptable because sump temperatures, pressures, and levels were not significantly impacted by the code of record change.

In SSIB-3-8, the NRC staff requested that the licensee provide information regarding changes that were included in the LAR UFSAR markups. Particularly, Table 6.3-1 of Attachment 3-4 of the August 20, 2015,[20] submittal shows the UFSAR changes for ECCS Component Parameters, specifically required and available NPSH. The NRC staff was concerned that the changes could be substantially non-conservative if they were not a result of changing the reference location on the pumps for calculation of $NPSH_R$. The licensee stated that the parameter values changed from the original plant design with sump screens primarily due to accounting for the strainer debris load, a change in the reference point for NPSH, a refinement of the containment water level calculation, and a redefinition of NPSH available to exclude the debris head loss and CSHL. The licensee stated that the reference point for the NPSH values listed in Table 6.3-1 were changed from the centerline of the pump suction nozzle to the pump first stage impeller. The NRC staff noted that the magnitude of the changes were driven by the redefinition of the NPSH reference point and found the remainder of the response acceptable. Therefore, the NRC staff finds the response to SSIB-3-8 is acceptable.

In SSIB-3-9, the NRC requested clarification of the term TSHL as used in the UFSAR markup that was included in the LAR. The licensee stated that the term was defined in the UFSAR markup. The NRC staff finds this response is acceptable since the term TSHL was defined.

UFSAR Section 6.2.1.3, "Mass and Energy Release Analyses for Postulated Loss of Coolant Accidents," states the use of WCAP-10325-P-A[80] methodology for LOCA mass and energy (M&E) release analysis. Westinghouse has issued Nuclear Safety Advisory Letter (NSAL)-06-6, "LOCA Mass and Energy Release Analysis,"[81] NSAL-11-5, "Westinghouse LOCA Mass and Energy Release Calculation Issues,"[82] and NSAL-14-2, "Westinghouse Loss-of-Coolant Accident Mass and Energy Release Calculation Issue for Steam Generator Tube Material Properties,"[83] and InfoGram (IG)-14-1, "Material Properties for Loss-of-Coolant Accident Mass and Energy Release Analyses,"[84] reporting errors in this methodology. Also, a new methodology using the GOTHIC code is used for LOCA containment analysis for which the M&E input needs to be corrected based on the above NSALs and the InfoGram. The licensee did not provide any information regarding the correction of errors in the current LOCA M&E release analysis in its August 20, 2015, supplement. In question SCBV-3-1, the NRC staff asked the licensee to provide information regarding the changes, and justification of changes, in inputs and assumptions from the current analysis, and results for the following revised licensing basis containment analysis: (a) LOCA containment M&E release analysis, (b) pressure and temperature response analysis for containment integrity, (c) peak temperature analysis for equipment environmental qualification, (d) sump temperature and level response for NPSH analysis, and (e) minimum containment pressure response for ECCS analysis.

In its July 18, 2016, response to items (a), (b), and (c) of SCVB-3-1,[25] the licensee referred to its March 17, 2016,[85] response to a similar question on the LAR for extension of the containment leakage rate testing program. In this response, the licensee stated that the revised M&E analysis was performed using the NRC-approved methodology WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version," May 1983, [86] after correction of issues reported in NSALs 06-6, 11-5, 14-2 and InfoGram IG-14-1. The correction resulted in an increase in the M&E release. The description in the August 20, 2015 supplement included a sump temperature and containment pressure that were calculated as part of the LOCA containment pressure and temperature response analysis using the CONTEMPT computer code. The revised analysis was performed using the NRC-approved methodology documented in Dominion topical report DOM-NAF-3, "GOTHIC Methodology for Analyzing the Response to Postulated Pipe Ruptures Inside Containment," September 2006, [87] with the GOTHIC computer code. The M&E correction for the issue reported in InfoGram IG-14-1 was incorporated by performing sensitivity analyses described in licensee's letter dated March 17, 2016, and accepted by the NRC staff in the April 29, 2016, SE. [88]

In its response to item (d) of SCVB-3-1, the licensee stated that for each break, two cases were analyzed; one case maximizes the containment pressure and the other maximizes the containment temperature following the input guidelines for containment pressure and temperature analysis provided in Table 3.6.1 of DOM-NAF-3. [87] The licensee selected the maximum sump temperature using the split heat transfer Option as described in Section 3.3.2 of DOM-NAF-3. The change from the use of CONTEMPT to GOTHIC resulted in a higher peak sump temperature which did not significantly affect the NPSH evaluation. The containment peak pressure and temperature response analysis using GOTHIC resulted in a sump temperature response which bounded the response using CONTEMPT. Therefore, an analysis for maximizing the sump temperature response using GOTHIC was not considered necessary. The NRC staff determined that the higher sump temperature response in the current CONTEMPT analysis results in a more limiting (lower) NPSH available at the suction inlet to the pumps that draw water from the sump during the recirculation phase of a LOCA. The licensee stated that density change for the sump water due to temperature did not have an adverse effect on NPSH calculations. There was no change to the sump level determination since the increase in sump temperature (and corresponding decrease in sump fluid density) did not have an adverse impact on NPSH available.

In its response to item (e) of SCVB-3-1, the licensee stated that the results of the post-LOCA minimum containment pressure response for ECCS analysis performed by Westinghouse are provided in UFSAR Section 6.2.1.5. The licensee further stated that Westinghouse has confirmed the NSALs 06-6, 11-5, and 4-2 and IG 14-1 do not impact the results presented in UFSAR Section 6.2.1.5. The NRC staff determined that the correction of errors reported in the NSALs and the InfoGram, which increases the M&E release into the containment, would only impact the limiting higher pressure and temperature response, so that a revised minimum pressure response with a higher M&E release would be bounded (higher) by the current minimum pressure obtained from lower values of M&E release.

In summary, in SCVB-3-1, the NRC staff requested that the licensee provide information regarding the M&E released to the containment during a LOCA. The NRC staff was concerned that the containment analysis did not account for vendor notices that described changes that should be made to the methodology. The licensee stated that a similar question had been asked regarding a different LAR that was under NRC review and referenced the response to the question. In addition, the licensee provided additional information specifically regarding how this issue affected the sump temperature and level calculations associated with NPSH calculations, and the containment pressure calculations. The licensee concluded that the calculations were performed acceptably. The NRC staff reviewed the response and finds it acceptable.

The NRC staff reviewed the licensee's evaluation against the NRC staff-accepted guidance and concludes that the licensee has appropriately validated that the plant design provides adequate margin between the NPSH available and the NPSH required for each pump taking suction from the recirculation sump. Therefore, the NRC staff concludes that the licensee's evaluation of NPSH is acceptable.

3.4.2.8.6 Chemical Effects

The initial STP risk-informed pilot approach considered head loss from chemical effects through the use of exponential probability functions developed for small break, medium break, and LBLOCAs. The CASA Grande head loss model first calculated conventional (i.e., from fibrous and particulate debris) head loss for a given pipe break and then applied a chemical bump-up factor (multiplier) that was independent of the conventional head loss. This approach relied on the use of engineering judgment to establish separate probability functions for the three break size categories. As part of the initial NRC review of the STP pilot, the NRC staff asked 22 RAIs related to chemical effects. STP provided responses to the NRC staff RAIs, including chemical effects, in three letters dated May 22, 2014,[15] June 25, 2014,[16] and July 15, 2014.[17] In the enclosure to the third set of responses to the NRC Round 1 chemical effects RAIs for the STP pilot, the licensee introduced a potential alternate method for determining chemical head loss. This alternate method used existing chemical effects head loss test results to calculate chemical head loss on a precipitate mass per sump strainer area basis. The chemical loading parameter in this alternate approach was termed, "L*". In contrast to the CASA Grande chemical effects bump-up multiplier, chemical effects head loss determined using the L* approach was added to the conventional head loss to determine a total head loss.

Subsequent to receiving the RAI Round 1 responses, the NRC staff audited the licensee's supporting documentation and discussed technical details of the pilot submittal with representatives from STP in Albuquerque, New Mexico, in September 2014.[89] The objective of the audit was to improve staff understanding of the STP pilot in a number of areas, including chemical effects. During this audit, representatives from STP stated that the chemical effects evaluation method was being changed from the original chemical bump-up multiplier discussed in Enclosure 4-3 of the licensee's letter dated November 13, 2013,[4] to the aforementioned L* chemical head loss factor.

On March 3, 2015,[90] RAI Round 2 was sent to the licensee. In the chemical effects area, the questions were intended to obtain clarification on some of the first round RAI responses and additional information concerning the new alternate chemical effects (L*) methodology. The licensee responded to the RAI Round 2 on March 25, 2015.[19]

The licensee's letter dated March 25, 2015, also provided a description of a significant revision to the overall risk-informed evaluation methodology used in the STP pilot license amendment. The revised RoverD analysis method is less complex compared to the initial STP risk-informed pilot approach.

The licensee's change in the overall risk-informed evaluation methodology to RoverD also resulted in a significant change to the chemical effects evaluation methodology. In the current RoverD chemical effects approach, the licensee is using results from strainer testing performed in 2008 (that included addition of chemical precipitates) to bound most pipe breaks in a deterministic manner. This change in the chemical effects evaluation method renders previous questions related to a risk-informed chemical effects approach moot. This includes questions related to the chemical bump-up factor, the L* approach, or questions related to chemical effects uncertainties (e.g., effects of radiation) for plants pursuing a risk-informed evaluation, rather than a deterministic chemical effects evaluation.

Since the RoverD chemical effects approach reverts back to a deterministic evaluation, the NRC staff RAI from 2009 based on the earlier sump strainer testing became relevant again. Accordingly, STP responded to the 2009 RAI with a letter dated August 20, 2015.[20] The remainder of this SE section addresses the current STP pilot deterministic approach to evaluating chemical effects.

The licensee evaluated chemical effects as part of the head loss testing that was performed at the Alden Laboratory in Holden, Massachusetts, during July 2008. The licensee calculated its plant-specific chemical precipitate load using the WCAP-16530-NP base methodology without refinements.[49] The licensee made a number of conservative assumptions when calculating the chemical precipitate amount. For example, the licensee assumed containment sprays would remain on for the entire 30-day ECCS mission time. The licensee also assumed that the post-LOCA sump temperature at the end of 2 days remained constant for the next 28 days instead of continuously cooling due to the residual heat removal heat exchangers. Assumptions of higher temperatures and continual wetting of plant materials from the containment sprays result in greater predicted amounts of chemical precipitate. In addition, the licensee used the maximum predicted precipitate amount for each precipitate independent of whether it occurred by assuming the maximum post-LOCA recirculation volume ($NaAlSi_3O_8$, $Ca_3(PO_4)_2$) or the minimum post-LOCA recirculation volume (AlOOH). With these assumptions, the STP plant-specific amounts of post-LOCA precipitate predicted by the WCAP-16530-NP methodology were 1,432 lbm $NaAlSi_3O_8$, 143 lbm AlOOH, and 359 lbm $Ca_3(PO_4)_2$. These STP plant-specific predicted quantities of chemical precipitates were then scaled to the Alden Laboratory strainer for head loss testing.

The STP ECCS sump strainer was designed by Performance Contracting, Inc. The strainer segment tested at Alden Laboratory was a full-size module identical to those installed in the STP containment, with a total surface area of 91.4 ft^2. The Alden facility included a test flume, pumps, test strainer module, instrumentation and controls, chemical mixing tanks, and associated piping needed to perform testing. The test flume measured 10-feet wide by 5-feet deep by 45-feet long. An overflow pipe maintained a constant water level during testing. Debris that flowed into the overflow pipe was captured by 10 micron bag filters that were periodically flushed to return the debris to the test flume. The test loop was initially heated to 120 °F using a recirculation line that contained heat exchangers. After the initial heat up, the heat exchangers were removed from service and immersion heaters were used to maintain the fluid temperature in the flume during testing.

During strainer testing, the debris addition was sequenced so that all fibrous and particulate debris was added before adding any chemical precipitates. The July 2008 STP test materials included: NUKON™ and Knauf Elevated Temperature fibrous insulation, microporous insulation, powdered Marinite® board, silica sand to simulate latent dirt, tin particulate (surrogate for zinc particulate), and acrylic coating powder. Particulate, followed by fibers, was added in batches such that the most transportable debris were introduced into the flume before the less transportable debris. Once all particulate and fibrous debris were introduced and a stable bed was formed on the test strainer, the test ran overnight with no further debris introductions prior to chemical precipitate addition.

Chemical precipitates were pre-mixed and placed in large holding tanks prior to addition to the test loop. After the precipitates were prepared, 1-hour settling tests were performed to ensure the chemical precipitates met the settlement criteria contained in WCAP-16530-NP-A. Due to concerns related to silicate hazards, STP substituted AlOOH precipitate for $NaAlSi_3O_8$ precipitate in its head loss testing. Chemical precipitates were added to the strainer test in 33 batches. The test vendor allowed one pool turnover between batches of AlOOH precipitate or if an AlOOH precipitate batch was followed by a $Ca_3(PO_4)_2$ precipitate batch. Two pool turnovers were completed between consecutive batches of $Ca_3(PO_4)_2$ precipitate. Results from the 2008 strainer tests showed that the addition of chemical precipitates caused an approximate doubling of the peak head loss relative to the conventional head loss (resulting from fibrous and particulate debris).

The licensee performed a chemical effects evaluation using the WCAP-16530-NP base method without refinements and with additional conservative assumptions. For example, the licensee assumed containment sprays will remain on for the entire 30-day ECCS mission time. As part of STP pilot's risk-informed analysis, the licensee determined the containment sprays will most likely be secured within 7 hours after a LOCA. The licensee also assumed contradictory (maximum and minimum) post-LOCA pool volumes to maximize the predicted amount of chemical precipitates. The overall chemical effects evaluation for STP is acceptable since the licensee used a previously reviewed methodology accepted by the NRC staff in the December 21, 2007, SE for WCAP-16530-NP-A.[49] The licensee's strainer test substituted AlOOH precipitates for the predicted $NaAlSi_3O_8$ precipitate. This substitution is acceptable to the NRC staff as discussed in the SE for WCAP-16530-NP-A.

The licensee performed sump strainer testing and simulated chemical effects by adding pre-mixed precipitate at the Alden Laboratory. The NRC staff is familiar with the licensee's test and evaluation methods for strainer testing because the staff has visited the vendor test facilities that performed testing for licensees and observed testing multiple times between 2005 and 2016. Summaries of various NRC staff visits are available in the Agencywide Documents Access and Management System (ADAMS).[91] In addition to these visits, NRC staff observed the July 2008 STP strainer test and determined that the testing methodology adhered to NRC staff guidance. The NRC staff's trip report for the July 2008 STP strainer test at the Alden Laboratory test facility is also available in ADAMS.[92] During that visit, NRC staff observed the preparation of chemical precipitates for STP strainer testing, and confirmed that the precipitates met the WCAP-16530-NP-A settlement conditions, including the more stringent settlement requirements detailed in the SE for head loss testing that credits near-field settling of debris.

As part of the NRC staff's review of the STP July 2008 strainer tests with chemical precipitates, the NRC staff asked question 52 related to transport of $Ca_3(PO_4)_2$ precipitate in the Alden test flume. The staff questioned whether $Ca_3(PO_4)_2$ settlement in the test flume upstream of the strainer would result in a non-representative amount of $Ca_3(PO_4)_2$ at the test strainer. The licensee responded to the question in the August 20, 2015, supplement.[20] The licensee subsequently provided clarification to the response in its October 20, 2016, supplement.[30] The NRC staff determined that the amount of $Ca_3(PO_4)_2$ that reached the test strainer was equal to or greater than the amount expected for the actual plant strainers, even if some precipitate settled in the test flume or was filtered by fibrous debris that had settled in the test flume upstream of the strainer. Thus, the NRC staff finds the licensee's response acceptable. The licensee's calculation for the amount of $Ca_3(PO_4)_2$ formed in the plant after a LOCA included Cal-Sil insulation as a source material. This material, however, has since been removed from containment, and the amount of $Ca_3(PO_4)_2$ precipitate added to the 2008 strainer test has additional margin on precipitate quantity relative to the current plant configuration. In addition, visual observation of the test flume showed that chemical precipitate remained suspended during the test, and post-test examination of the debris bed showed the bed thickness was mostly comprised of chemical precipitate.

Subsequent to the July 2008 strainer head loss testing and corresponding chemical effects evaluation, the licensee also performed extensive additional chemical head loss experiments (CHLEs) [10] at the University of New Mexico in support of the risk-informed resolution approach that was eventually superseded by the current RoverD approach. Although the NRC staff is not relying on the CHLE test results, the CHLE suite of tests provides additional evidence that the STP plant-specific chemical effects would be much less severe than those simulated in the 2008 strainer testing.

In summary, the NRC staff concludes that STP's evaluation of chemical effects is acceptable since the licensee used the WCAP-16530-NP base model methodology to calculate the precipitate quantity and to prepare chemical precipitates for the strainer head loss testing performed in 2008. The NRC staff's acceptance of this methodology for chemical effects evaluations is documented in the December 21, 2007, SE for WCAP-16530-NP-A. [93] The NRC staff observed the July 2008 STP strainer testing and concludes that there is reasonable assurance that the testing was performed according to NRC staff guidance. Finally, the licensee also adequately addressed all NRC staff chemical effects-related RAIs associated with testing conducted in July 2008.

3.4.2.8.7 Downstream Effects-Components and Systems

The licensee stated that close-tolerance subcomponents in pumps, valves, and other ECCS and CSS components were evaluated for potential plugging or excessive wear due to extended post-accident operation with debris-laden fluids. The licensee used WCAP-16406-P-A, Revision 1, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," March 2008, [50] and the accompanying NRC December 20, 2007, SE to perform the evaluations. [94] The licensee stated that no exceptions were taken to the WCAP-16406-P-A methodology.

Based on use of the approved methodology, the licensee stated that the following ECCS and CSS components evaluated for STP can accommodate sump bypass particles without blockage: throttle valves; pipes, valves, and instrumentation; orifices and eductors; heat exchangers; and nozzles. The licensee reviewed drawings and documents and found that the ECCS and CSS valves are not throttled. The licensee determined that, according to the criteria established in WCAP-16406-P-A, the wear impact on the valves was non-critical and no further erosion evaluation was warranted.

For pumps, the licensee evaluated the effects of debris ingestion through the sump screen on hydraulic performance, mechanical shaft seal assembly performance, and mechanical performance (vibration) of the pumps. The pumps identified for evaluation were the HHSI, LHSI, and CSS pumps. According to the methodology established in WCAP-16406-P-A, no effect on their hydraulic performance is expected. The licensee stated that the mechanical shaft seal assembly performance evaluation resulted in a recommendation to replace the LHSI, HHSI and CSS pumps' carbon/graphite packing assemblies with a more wear-resistant material. However, because STP has an atmospheric filtration system for the building where the pumps are located, replacement of the carbon/graphite seal bushing is not required because any discharge from the packing will not result in unfiltered releases.

The licensee stated that tube failure for heat exchangers will occur if the wall thickness after erosion is less than the required wall thickness to retain internal and external pressures. According to the methodology established in WCAP-16406-P-A, the minimum wall thickness required to retain both internal and external pressures is less than the resultant wall thickness after erosion. Therefore, the heat exchangers are not expected to fail.

WCAP-16406-P-A states that if the inside diameter of an orifice is increased by less than 3 percent due to erosive wear, the effect on system performance is negligible. For STP, the inside diameters of all the orifices are predicted to change by less than 3 percent and are therefore not expected to fail.

According to WCAP-16406-P-A, failure due to erosive wear for spray nozzles is expected to occur when the flow from the nozzle is increased by 10 percent due to the increase in the nozzle inner diameter. The licensee calculated the flow based on nozzle diameters before and after erosive wear. It was found that the flow is changed by less than 2 percent, which is less than the 10 percent limit; therefore, the nozzles are not predicted to fail.

The potential for blockage of the reactor vessel water level system is not evaluated since STP has a Westinghouse design reactor vessel water level system, for which WCAP-16406-P states there is no blockage concern due to the debris ingested through the sump screen during recirculation.

The licensee stated that no design or operational changes were made as a result of the current downstream evaluations.

The NRC staff reviewed the evaluation methods, the evaluation results, and the licensee's conclusions and concluded that the licensee followed the NRC staff-accepted guidance contained in TR-WCAP-16406-P-A, Revision 1, including the NRC SE for that document. The NRC staff concludes that the licensee performed an adequate downstream effects evaluation of components and systems and that the components are capable of performing their safety-related design functions for their required mission times after a LOCA because they used staff approved guidance for the evaluation and determined that wear effects would not adversely impact the plant response following a LOCA.

3.4.2.8.8 Downstream Effects-Fuel and Vessel

This section on downstream effects for the fuel and reactor vessel is also referred to as in-vessel or in-core thermal-hydraulic impacts. For the in-vessel thermal-hydraulic impacts, the licensee evaluated hot-leg break and cold-leg break scenarios using different methods. The table below is a summary of the different methods.

	Hot-leg Break	**Cold-leg Break**
Small Break	Deterministic Analysis (SE Attachment 2)	RoverD (Main SE)
Medium Break	Deterministic Analysis (SE Attachment 2)	RoverD (Main SE)
Large Break (> 16")	Risk-informed Analyses (Main SE)	RoverD (Main SE)

For small and medium hot-leg breaks, the licensee used thermal-hydraulic analyses to show that long-term core cooling would be maintained, even if the core inlet is blocked completely by debris. The NRC staff's review of the evaluation methodology and simulation results are in SE Attachment 2, "In-Vessel Thermal-Hydraulic Analysis."

For large hot-leg breaks, in order to simplify its thermal-hydraulic analysis, the licensee assumed that core damage would result for any hot-leg break greater than 16 inches. Not all large hot-leg breaks were shown to fail in the analysis, however, the licensee chose to risk-inform all large hot-leg breaks.

For the cold-leg breaks, the licensee determined the amount of debris that could reach the core inlet under bounding conditions, and compared the calculated amount to the threshold previously determined to be acceptable by the NRC staff. The licensee determined that the maximum amount of fiber that could reach the core is 7 grams per fuel assembly (gm/FA) under the limiting pump operating combination. All other cases result in a lower fiber load in the core.

The NRC staff's evaluation of STPNOC's methodology for calculating fibrous debris amounts is discussed in SE Section 3.4.2.7, "Debris Transport," and was found to be acceptable. The cited amount meets the metrics approved by WCAP-16793 and the associated NRC staff SE dated April 8, 2013.[44] The NRC staff finds that, under realistic conditions, the STP core fibrous debris load for cold-leg breaks will be less than 7 gm/FA, and is acceptable.

The licensee also stated that industry testing, accepted by the NRC, demonstrated that if the amount of fiber in the core is limited to 15 grams per fuel assembly, core cooling would not be compromised (WCAP-16793-NP-A, Revision 2, "Evaluation of Long Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," July 2013).[44] The licensee stated that the debris loading at STP following a cold-leg break was very small, and therefore boric acid precipitation (BAP) would not be affected by the debris.

The NRC staff notes that the 15 gm/FA limit was intended as a guideline for allowing licensees to address BAP at a later time through other industry initiatives, as discussed in the SE for WCAP-16793. Therefore, the NRC staff disagrees with the licensee's assertion that BAP is not impacted by the calculated debris loading at STP. This is because the NRC staff's conclusions in the SE related to WCAP-16793, do not specify a debris amount acceptable considering the effects of BAP following a cold-leg break. However, the NRC staff recognized in the SE for WCAP-16793 that very small amounts of debris have a low likelihood of impacting BAP and therefore the industry could address the issue at a later time.

Although the NRC staff concludes that the licensee did not address the effects of debris on BAP following a cold-leg break, the current STPNOC analysis of record for BAP is not likely to be adversely affected by the small quantities debris predicted to reach the core following a cold-leg break. Therefore, the NRC staff concludes that BAP will not be adversely impacted by approval of this amendment at STP, and STPNOC will resolve BAP for cold-leg breaks considering the effects of debris as part of the industry effort.

The NRC staff reviewed the licensee long-term core cooling evaluation model and the associated simulations for small and medium hot-leg breaks in SE Attachment 2. The NRC staff concludes that the long-term core cooling and associated simulations are acceptable for analyzing the impacts of in-vessel debris on the core and that there is reasonable assurance that the licensee addressed downstream, in-vessel effects for both hot-leg medium and small breaks. For a detailed discussion of the staff's review, see SE Attachment 2.

NRC Staff Conclusion Regarding Impact of Debris

Each of the aspects of the impact of debris area has been evaluated above. The NRC staff concludes that the sub-areas of upstream effects, screen modification package, sump structural analysis, NPSH, head loss and vortexing, chemical effects, downstream effects – components and systems, and downstream effects – fuel and vessel were adequately addressed. Based on the evaluations for each of these subsections, the NRC staff concludes that the impact of debris evaluation is acceptable.

3.4.2.9 Sub-model Integration

STPNOC used separate models to analyze various phenomena associated with debris effects. These models were used individually to evaluate each aspect of the scenario. Instead of calculating a head loss or in-vessel debris amount for each potential break location, the licensee used the results of tests to determine pass-fail criteria and compared these criteria to the results of analyses that calculated the amounts of debris that could be present at the strainer or in the core. This is a simplified, bounding approach that does not need sub-model integration.

For the in-vessel evaluation for hot-leg breaks less than 16 inches, the licensee performed a thermal-hydraulic analysis that assumed that the core inlet was completely blocked. This analysis showed that, for hot-leg breaks 16 inches and below and cold-leg breaks, long-term core cooling acceptance criteria was maintained (see NRC staff evaluation in SE Attachment 2). For breaks greater than 16 inches in hot-leg piping, in order to simplify the analysis, the licensee assumed that core damage would occur and assigned a corresponding risk value in the risk-informed evaluation.

3.4.2.10 Systematic Risk Assessment

RG 1.174[41] states that the licensee may use its risk assessment to address the principle that proposed increases in risk are small and are consistent with the intent of the NRC's Safety Goal Policy Statement.

In its October 20, 2016, letter,[30] the licensee described its approach used to quantify the risk (CDF and LERF) attributable to debris. This risk was defined as the difference in risk between the as-built, as-operated plant (with debris) and a hypothetical plant with no risk from debris.

As discussed in SE Section 3.4.2.1, "Scope of the Systematic Risk Assessment," the licensee first screened out or bounded the risk attributable to debris for all plant operating modes except full-power operation and for all initiating events except medium- and large-break LOCAs. A second, location-specific screening process further refined the scope to a discrete set of primary piping welds by comparing the amount and type of debris generated by each break to a threshold value (determined by testing). For the purposes of risk quantification, any break predicted to generate and transport debris in excess of the tested value (referred to as a "critical" break) was assigned a conditional core damage probability of 1.0. Breaks predicted to produce debris below the threshold value were assigned a conditional core damage probability of 0. The licensee provided a list of all 628 welds in the STP RCS. This list showed the amount of debris that could be produced assuming a DEGB occurred at each weld. For Case 1 (two or three ECCS and CSS trains available), 45 welds were determined to be critical. For Case 2 (one ECCS and CSS train available), a greater number of welds (95) were determined to be critical since for these scenarios only one-half of the amount of debris was necessary to cause failure.

The licensee identified uncertainties in the results of the core thermal-hydraulic analysis of certain RCS hot-leg breaks. Specifically, adequate core cooling could not be demonstrated with adequate margin for RCS hot-leg breaks greater than 16 inches. The licensee identified 20 hot-leg welds that can support breaks greater than 16 inches. Of those 20, a total of 12 were already included in the list of 45 and 14 were included in the list of 95. The final list of critical welds included 53 welds for the Case 1 and 101 welds for Case 2. The remaining welds (i.e., those welds not on these lists) were screened from further consideration.

The next step reported by the licensee was to estimate the weld-specific break frequency for each of the aforementioned screened-in welds. In order to perform this step, the licensee made several key assumptions:

1. The frequency of a LOCA at a given location is a function only of the break size (e.g., all 7-inch breaks have the same frequency, all 8-inch breaks have the same frequency, etc.). This approach is sometimes referred to as the "top down" approach because it starts with a plant-wide ("top level") LOCA frequency and allocates it evenly to various break locations according to break size.

2. The licensee also considered the question of how to model complete versus partial breaks. To do this, the licensee presented two sets of results. One set used the "continuum break" assumption that a complete break of a given size in one pipe is exactly as likely as a partial break of the same size in a larger pipe. The other set of results made a "DEGB-only" assumption, meaning that only complete, DEGBs were evaluated.

3. Finally, the licensee presented results using both the geometric and arithmetic mean aggregation schemes described in NUREG-1829.[63]

In employing the "continuum break" assumption (key assumption 2), the licensee determined the smallest break size at each critical weld that would just result in exceeding the debris threshold. To do this, the licensee performed a sweep using 1-degree angular intervals and 0.01-inch size intervals at each critical weld location. Once the smallest critical break size was identified, all breaks this size or larger were assumed to be critical as well (i.e., assigned a conditional core damage probability of 1.0). This approach is conservative because the NUREG-1829 LOCA frequencies are non-increasing as break size increases, so that a smaller break would have a higher initiating event frequency. However, the licensee also divided the resulting frequency by the number of welds "capable" of having a LOCA of that smallest size or larger. A smaller break size can mean that there are more "capable" welds, resulting in a decrease in the LOCA frequency assigned to a given weld. Whether this assumption is conservative or non-conservative is a fundamental uncertainty associated with the continuum break assumption and is discussed in SE Section 3.4.2.11, "Sensitivity and Uncertainty Analyses."

For each set of assumptions, the licensee stated that delta core damage frequency (ΔCDF) was determined for two plant operating configurations: Case 1 and Case 2. The ΔCDF attributable to debris was reported as the weighted average of the ΔCDF for Case 1 and Case 2, which were denoted as ΔCDF_1 and ΔCDF_2, respectively, in the discussion that follows.

The total CDF attributable to debris, $\hat{\phi}$, was computed as

$$\hat{\phi} = \omega_1 \Delta CDF_1 + \omega_2 \Delta CDF_2$$

The licensee used its plant-specific PRA model to estimate the weight factors ω_1 and ω_2, which represent the relative frequency of LBLOCA scenarios for Case 1 and Case 2. In its October 20, 2016, letter, the licensee stated, in part, that:[§]

> In referring to the number of trains of ECCS pumps operating following a LLOCA [LBLOCA], it is important to consider all the applicable pump types; that is HHSI, LHSI, and containment spray pumps. Since the spray pumps are eventually shut down after swap over to recirculation and the HHSI pumps are of much lower capacity, the frequencies of ECCS trains operating is evaluated on the status of the LHSI pumps only. Therefore f_1 (for single train operation) is computed as the sum of all pump state frequencies in which exactly one LHSI pump is operating, regardless of the status of the HHSI or containment spray pumps. Similarly, f_2 (that is for two or more trains in operation) is based on the sum of all pump states in which 2 or 3 trains of LHSI pumps are operating, again, regardless of the number of HHSI pumps or containment spray pumps operating. For both computations, only those pump states in which the large break LOCA is successfully mitigated in the absence of GSI-191 phenomena are tracked. There are no success sequences in which zero LHSI pumps operate.

The licensee stated that the annual frequencies for the two plant states discussed above are 4.16E-6 (f_2) and 1.55E-9 (f_1) for Case 1 and Case 2, respectively. This implies that two or three sumps are expected to be in operation following a LBLOCA about 99.96 percent (ω_1) of the time, and in single-sump operation only about 0.04 percent (ω_2) of the time.

The licensee reported the following risk estimates, in CDF per year:

Continuum Break Model						
Quantile	Case 1 GM	Case 1 AM	Case 2 GM	Case 2 AM	$\hat{\Phi}$ (GM)	$\hat{\Phi}$ (AM)
5^{th}	3.30E-10	7.88E-09	4.41E-09	2.68E-08	3.31E-10	7.89E-09
50^{th}	9.44E-09	2.05E-07	9.92E-08	5.50E-07	9.47E-09	2.05E-07
95^{th}	4.38E-07	5.78E-06	2.15E-06	1.36E-05	4.39E-07	5.79E-06
Mean	1.50E-07	1.90E-06	5.90E-07	4.31E-06	1.50E-07	1.90E-06
DEGB-Only Model						
5^{th}	3.47E-10	7.78E-09	3.35E-09	2.23E-08	3.49E-10	7.78E-09
50^{th}	8.62E-09	1.92E-07	7.60E-08	4.72E-07	8.64E-09	1.92E-07
95^{th}	2.84E-07	6.02E-06	1.72E-06	1.19E-05	2.84E-07	6.03E-06
Mean	9.02E-08	1.82E-06	4.82E-07	3.80E-06	9.03E-08	1.82E-06

§ Note that the above discussion uses the subscript 1 on ω to refer to Case 1 (two or three trains operating) whereas the subscript 1 on *f* refers to Case 2 (single-train operation).

The licensee also estimated the increase in LERF attributable to debris. The licensee noted that only LBLOCA events remained screened into the risk assessment of debris. The licensee used its plant-specific PRA model to calculate the base CDF and LERF for LBLOCA scenarios, assuming that sump recirculation failed due to strainer plugging. The resulting conditional probability of LERF given core damage under these circumstances was found to be 2.5E-3. The licensee used this fraction to estimate the delta LERF (ΔLERF) attributable to debris and concluded that the RG 1.174 acceptance guidelines were met for ΔLERF.

With respect to the licensee's choice of a "top down" allocation scheme (key assumption 1 above), the NRC staff notes that there is a lack of consensus** regarding how to apportion plant-wide LOCA frequencies to individual weld locations. NUREG-1829 contains only qualitative statements about similarly sized welds in different locations having different expected rupture frequencies due to degradation mechanisms (e.g., hot-leg versus cold-leg).

Similarly, the NRC staff notes that there is a lack of consensus on how to compare the likelihood of a complete break of a given size to a partial break of the same size (key assumption 2). NUREG-1829 states that the expert panel generally felt that a complete break of a given size is more likely than a partial break of the same size but no quantitative information is stated.

NUREG-1829 does not endorse a specific aggregation scheme (key assumption 3), but states that "the purposes and context of the application must be considered when determining the appropriateness of any set of elicitation results."

With respect to key assumptions 2 and 3, the NRC staff finds that the licensee's reported results are acceptable because the licensee considered alternative approaches and in all cases, the reported results were within the RG 1.174 acceptance guidelines.

To assess the impact of key assumption 1, the NRC staff performed a conservative confirmatory calculation. This calculation assumed core damage for the smallest critical break and all breaks (regardless of location) equal to or larger than this break. This confirmatory calculation is described below.

The licensee reported that the smallest LOCA sizes predicted to generate and transport debris in excess of the tested amount are 12.814 inches for Case 1 and 9.34 inches for Case 2. The NRC staff used log-linear interpolation to determine the plant-wide LOCA frequencies for a break of each size or larger. Log-linear interpolation is judged acceptable because the pipe break sizes span approximately one order of magnitude and the non-exceedance frequencies span several orders of magnitude. Interpolating the LOCA frequencies from NUREG-1829,

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** RG 1.174 defines a "consensus" model as one that the NRC has utilized or has accepted for the specific risk-informed application under consideration or that has a publically available basis, has been peer reviewed, and has been widely adopted by the appropriate stakeholder group.

Table 7.13 (arithmetic mean aggregation) and from Table 7.19 (geometric mean aggregation) yields the following estimates of CDF.

NRC Staff Confirmatory Calculation Using Bounding Approach (per year)						
Quantile	Case 1 GM	Case 1 AM	Case 2 GM	Case 2 AM	$\hat{\phi}$(GM)	$\hat{\phi}$(AM)
5th	7.46E-10	1.38E-08	4.30E-09	3.61E-08	7.47E-10	1.38E-08
50th	2.08E-08	3.42E-07	1.50E-07	7.59E-07	2.08E-08	3.42E-07
95th	8.64E-07	9.17E-06	2.78E-06	1.86E-05	8.65E-07	9.17E-06
Mean	2.84E-07	3.02E-06	7.98E-07	5.96E-06	**2.85E-07**	**3.03E-06**

GM=geometric mean
AM=arithmetic mean

Note that this bounding approach, which assigns the interpolated plant-wide LOCA frequency from NUREG-1829 for the smallest break size capable of resulting in failure of the strainers, addresses both the welds that exceeded the debris amount acceptance criteria and the hot-leg welds that were added because of uncertainty regarding the thermal-hydraulic calculations.

Applying the licensee-derived ratio of LERF to CDF presented above (2.5E-3) to the mean-value results of the NRC staff's confirmatory calculation (shown in bold font in the above table) yields an increase in mean-value LERF attributable to debris of 7.12E-10 using geometric mean and 7.56E-9 using arithmetic mean aggregation.

The NRC staff concludes that the approach the licensee used to estimate the risk attributable to debris is acceptable for several reasons. First, the initiating event frequencies for critical welds were determined using NUREG-1829 and core damage was conservatively assumed to occur anytime a LOCA of sufficient size occurred at one of those welds. Further, the licensee investigated how several key assumptions could influence the results of the risk analysis by performing sensitivity studies. Finally, the NRC's confirmatory calculation provides an upper-bound estimate on the risk assuming that the NUREG-1829 LOCA frequencies are taken as consensus values.

3.4.2.11 Sensitivity and Uncertainty Analyses

RG 1.174 states, in part, that "[t]he scope, level of detail, and technical adequacy of the engineering analyses conducted to justify any proposed [licensing basis (LB)] change should be appropriate for the nature and scope of the proposed change. The licensee should appropriately consider uncertainty in the analysis and interpretation of findings." Consistent with RG 1.174, comparisons to the risk acceptance guidelines should be made with appropriate consideration of the uncertainties involved. The fundamental objective of an uncertainty evaluation is to provide confidence that the risk acceptance guidelines are met.

In its letter dated August 20, 2015,[20] the licensee provided information relevant to uncertainty. Although the context of that analysis in the licensee's submittal was "safety margins," the NRC staff regards such analysis to be pertinent to uncertainty analysis, because the licensee addressed effects of uncertainty in the ΔCDF and ΔLERF. RG 1.174 refers to the uncertainty framework described in NUREG-1855, Volume 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisions Making, Main Report," March 2009,[95] for acceptable approaches to addressing uncertainty. Section 7 of NUREG-1855 states that the goal of an uncertainty analysis is examining challenges posed by uncertain elements of the risk-informed analysis to the conclusion that ΔCDF and ΔLERF estimates are within acceptance guidelines defined in RG 1.174.

The licensee's conclusion that ΔCDF and ΔLERF attributable to debris is acceptably small (i.e., within RG 1.174 acceptance guidelines) depends on the following key aspects:

1. Strainer tests with bounding amounts of all debris types indicate that strainers would withstand any debris load for most debris types, except for fiber.

2. Analyses used to assess the effect of debris on long-term core cooling included appropriate conservatisms.

3. Sensitivity analyses regarding LOCA frequency allocation and conservative assumptions with respect to break orientations.

With respect to the uncertainty associated with key aspect 1, the NRC staff determined that the licensee conducted a test using realistic or conservative inputs and assumptions. Additional details that describe the acceptability of the testing are discussed in the head loss testing and are provided in SE Sections 3.4.2.8, "Impact of Debris," and 3.4.2.8.3, "Head Loss and Vortexing." The test inputs were developed and the test was performed using NRC staff guidance that is intended to ensure conservative results such that the NRC is confident that the test will result in a head loss that is bounding for the plant-specific conditions. The amounts of debris included in the test, with the exception of fiber, represented the maximum amount of debris that could arrive at the strainer following any break. Most locations would result in lesser amounts of debris. Even though it was later determined that larger amounts of some particulate debris types could be generated and transported, the test method resulted in significant conservatism in total particulate debris amounts for most locations and every location is bounded. The debris amounts and characteristics used in the test are discussed in SE Section 3.4.2.6, "Debris Source Term." Although the licensee used some assumptions in the development of these inputs that were non-conservative with respect to accepted staff guidance, the licensee was able to justify that the assumptions were acceptable. These issues are discussed in the "Debris Source Term" section of this SE. The NRC staff evaluated the licensee's plant-specific testing to verify that conservative inputs were used to develop the strainer head loss testing results. The NRC staff concludes the inputs are adequate and the testing results provide reasonable assurance that uncertainty has been addressed and that the head loss associated with the debris load in the testing bounds any head loss that would occur in the plant for the same debris load.

The NRC staff determined that the type and amounts of chemical precipitates included in the July 2008 strainer tests were prepared following the WCAP-16530-NP-A protocol that has been reviewed and accepted by the NRC staff.[49] The NRC staff's December 21, 2007, SE for WCAP-16530-NP-A found the conservative assumptions in the methodology accounted for uncertainties in the actual formation of chemical products in the post-LOCA environment. In addition, the licensee performed plant-specific CHLE at the University of New Mexico that provided further evidence that STP plant-specific chemical effects simulated in the 2008 strainer testing were bounding. Thus, head loss measured in the STP strainer experiments is expected to be higher than in actual LOCA conditions. The NRC staff concludes that the uncertainty analysis is adequate, because July 2008 testing amounts of chemical precipitates have been demonstrated to be bounding. Therefore, uncertainties in chemical precipitates are not expected to challenge the conclusion that ΔCDF and ΔLERF are within RG 1.174 acceptance limits.

The licensee claimed that the July 2008 strainer tests were performed loading debris in an order intended to avoid underestimating strainer head losses. In actual LOCA conditions, the licensee argued that debris beds would not be as uniform as in the tests, and cause smaller pressure loss. The NRC staff finds the licensee's arguments reasonable, supporting the notion that a strainer system under attainable LOCA conditions would experience lower head loss than in the July 2008 strainer tests. Therefore, uncertainties in the debris load order and configuration are not expected to challenge the conclusion that ΔCDF and ΔLERF are within RG 1.174 acceptance limits.

The NRC staff evaluated the licensee's plant-specific testing to verify that conservative inputs were used to develop the strainer head loss testing results. The NRC staff concludes the inputs are adequate and the testing results provide reasonable assurance that uncertainty has been addressed and that the head loss associated with the debris load in the testing bounds any head loss that would occur in the plant for the same debris load.

With respect to key aspect 2, two scenarios were considered: the cold-leg break and the hot-leg break. Each of these was treated differently by the licensee.

For the cold-leg break scenario, the licensee calculated the amounts of debris that may reach the core under several scenarios including those that are very unlikely, beyond the plant design basis, and limiting with respect to the amount of debris. Even under these limiting scenarios, the debris amounts are small compared to amounts shown to allow acceptable cooling flow by industry testing. Although these debris amounts were not found acceptable by the NRC staff, they were evaluated during the staff's review of WCAP-16793. [44] In its review of the WCAP, the staff accepted a hot-leg break limit of 15 grams per fuel assembly with the understanding that a cold-leg break could potentially result in about one half of this quantity of debris reaching the core. This understanding allowed the staff to postpone the requirement for plants to explicitly evaluate the effects of debris on BAP because the staff concluded that such a small amount of debris was unlikely to affect the assumptions in BAP evaluations. The tests associated with this topical report indicated that adequate flow to the core could be maintained for a cold-leg break if 18 grams of fiber per fuel assembly or less reached the core. The staff did not approve this amount, but notes that it provides a metric for comparison. The licensee calculated that under realistic cold-leg break scenarios, about 2 grams of fiber per fuel assembly could reach the core. Under limiting conditions, the licensee calculated about 7 grams per fuel assembly. These values are within those acceptable by the staff in the April 8, 2013, SE on WCAP-16793. In addition, they are significantly below the 18 grams discussed above.

For the cold-leg break, the potential for BAP is an important consideration for long-term core cooling. As discussed in Section 3.4.2.8, "Impact of Debris," the NRC staff did not conclude that the licensee's evaluation of BAP was acceptable, but based on the small debris load calculated to reach the reactor following a cold-leg break, concluded that it does not pose a safety concern and may be addressed at a later time.

The NRC staff concludes that uncertainties associated with the effects of fiber on core cooling for the cold-leg break are not expected to challenge the conclusions of the RoverD evaluation because they contain adequate margin between the amounts of debris calculated to reach the core and those shown to allow adequate flow. Sensitivity studies indicate that the likely amounts of debris reaching the core for cold-leg breaks is very small.

For the hot-leg break in-vessel evaluation, the NRC staff concludes that the thermal-hydraulic analysis was performed using significantly conservative inputs and assumptions so that uncertainties associated with that analysis will not challenge its conclusions. Attachment 2 to this SE discusses the uncertainties in the hot-leg break analysis.

With respect to key aspect 3, the licensee quantified the conditional probability for a break to cause a strainer failure event by determining whether a break would produce debris in excess of the threshold amount determined by the testing. As discussed in SE Section 3.4.2.10, "Systematic Risk Assessment," the frequency of such a break was considered equal to the core damage frequency attributable to debris generated by that break. The licensee summed all such break locations to determine the increase in CDF from debris.

There are several conservatisms in this approach. The conservative nature of the test that set the threshold for strainer failure is stated above. The licensee's determination of the smallest break size that would just exceed the threshold considered the worst-case orientation of the break. That is, the licensee overestimated the conditional failure probability by accounting solely for break orientations producing maximum debris amounts. In many cases, other break orientations could produce much less debris. This results in a conservative list of weld locations that can produce debris sufficient to exceed the acceptance criteria determined by the testing. In addition, finding the smallest break size, by determining the most conservative jet orientation, at each location on this list is also a conservative approach because smaller size breaks are considered more likely and therefore lead to higher CDF and LERF estimates.

In its letter dated May 22, 2014, [15] the licensee stated that welds on pipes are proper representatives of LOCA break sites. Non-pipe-break LOCAs (e.g., valve bodies and manways, etc.) were considered by the licensee in the following manner: (1) non-pipe locations that could produce a LOCA were identified; (2) the quantity and type of debris that could be generated at these locations was qualitatively compared to nearby pipe welds; and (3) the licensee concluded that pipe weld locations bounded non-pipe locations. Therefore, the licensee used the plant-wide LOCA frequencies from NUREG-1829, which includes both pipe and non-pipe locations, for the pipe-break LOCAs. An additional assumption is that pipe-break LOCA events only occur at welds; this is consistent with the guidance in NUREG-1829.

Furthermore, the licensee equated exceedance of the threshold fiber amounts (191.78 lbm of fiber fines for two or three operating trains of the ECCS, and 95.89 lbm for one operating train) with strainer failure, and strainer failure with core damage. The licensee noted that there are margins included in the determination of debris loading acceptance criteria through testing. In addition, strainer failure may not result in core damage in every case, because the licensee has procedures to detect and mitigate strainer blockage even during scenarios where ECCS and CSS performance is challenged by debris. However, this potential conservatism may be small, because a DEGB at some of the welds produces several times the debris acceptance criteria amount.

The LOCA frequency is a key uncertainty of the analysis; however, as stated above, the licensee used NUREG-1829 as the source of LOCA frequency information and presented results based on several different sets of assumptions.

The NRC staff concludes that the licensee appropriately identified sources of uncertainty and, consistent with RG 1.174 addressed them with conservative assumptions, qualitative arguments, or sensitivity studies.

The only source of uncertainty not addressed in this manner was the "top down" approach to LOCA frequency allocation; however, the NRC staff's confirmatory calculation of risk attributable to debris, presented in the Systematic Risk Assessment section, bounds the uncertainty associated with the top down approach.

3.4.3 NRC Staff Conclusion Regarding the Increase in Risk

Principle 4 in RG 1.174 states that any increase in risk associated with a proposed change should be small (i.e., within the risk acceptance guidelines). The risk acceptance guidelines are presented in Figures 3 and 4 in RG 1.174 for CDF and LERF, respectively. Note that those figures use the mean values of CDF, ΔCDF, LERF, and ΔLERF.

As stated in SE Section 3.4.1, "NRC Review of the Base PRA Model," the STP base CDF for each unit was reported as 7.9E-6 per year and the base LERF was 4.7E-7 per year. From RG 1.174 Figure 3, this base CDF means that an increase in CDF of 1E-5 per year or less is considered "small" and an increase of 1E-6 per year or less is "very small." Similarly, from Figure 4, the base LERF means that an increase in LERF of 1E-6 per year or less is considered "small" and an increase of 1E-7 per year or less is "very small."

The licensee's mean-value estimates for the increase in CDF attributable to debris provided in its letter dated October 20, 2016,[30] and the corresponding conclusion from RG 1.174, Revision 2, Figure 4 are shown in the following table, which also includes the results of the NRC staff's confirmatory calculation:

Model	Aggregation	Mean ΔCDF (per year)	Increase
Continuum	GM	1.50E-7	Very small
Continuum	AM	1.90E-6	Small
DEGB only	GM	9.03E-8	Very small
DEGB only	AM	1.82E-6	Small
NRC Confirmatory	GM	2.85E-7	Very small
NRC Confirmatory	AM	3.03E-6	Small

GM=geometric mean
AM=arithmetic mean

The base PRA risk and various estimates in the increase in CDF associated with debris were found to meet the acceptance guidelines in RG 1.174 including consideration of uncertainty. The acceptance guidelines for LERF (Figure 5 of RG 1.174) are an order of magnitude lower than CDF for both base LERF and ΔLERF. The licensee determined the ratio of CDF to LERF for the scenarios of interest to be 2.5E-3. The increase in LERF is therefore "very small" for all the cases shown in the above table. Therefore, the NRC staff concludes that the licensee adequately demonstrated that the risk attributable to debris is acceptable because:

- The licensee used a PRA of the appropriate scope, level of detail, and technical adequacy
- The risk-informed approach used by the licensee to address the effects of debris on long-term core cooling is acceptable
- The increase in risk meets the risk acceptance guidelines as defined RG 1.174.

3.5 Key Principle 5: The Impact of the Proposed Change Should Be Monitored Using Performance Measurement Strategies

RG 1.174,[41] Section C.3, "Element 3: Define Implementation and Monitoring Program," states, in part, that:

> The primary goal of Element 3 is to ensure that no unexpected adverse safety degradation occurs due to the change(s) to the LB [licensing basis]. The [NRC] staff's principal concern is the possibility that the aggregate impact of changes that affect a large class of SSCs could lead to an unacceptable increase in the number of failures from unanticipated degradation, including possible increases in common cause mechanisms. Therefore, an implementation and monitoring plan should be developed to ensure that the engineering evaluation conducted to examine the impact of the proposed changes continues to reflect the actual reliability and availability of SSCs that have been evaluated. This will ensure that the conclusions that have been drawn from the evaluation remain valid.

In its letter dated October 20, 2016,[30] the licensee stated that it has implemented programs and procedures to evaluate and control potential sources of debris in containment, including Technical Specification Surveillance Requirements that require visual inspections of all accessible areas of the containment to check for loose debris, and each containment sump to check for debris. The licensee stated that its design change control procedure includes provisions for managing potential debris sources such as insulation, qualified coatings, addition of aluminum or zinc, and potential effects of post-LOCA debris on recirculation flow paths and downstream components. The procedure has been augmented to explicitly require changes that involve any work or activity inside the containment be evaluated for the potential to affect the following:

- Reactor coolant pressure boundary integrity
- Accident or post-accident equipment inside containment
- Quantity of metal inside containment
- Quantity or type of coatings inside containment
- Thermal insulation changed or added
- Post-LOCA recirculation flow paths to the emergency sumps
- Post-LOCA recirculation debris impact on internals of fluid components
- Addition or deletion of cable

The licensee stated that a 10 CFR 50.59 screening or evaluation is required to be completed for all design changes, which ensures that new insulation material that may differ from the initial design is evaluated for GSI-191 concerns. It also stated that it has implemented procedures to ensure that Service Level 1 protective coatings used inside containment are procured, applied, and maintained in compliance with applicable regulatory requirements. The licensee noted that the 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," also known as the Maintenance Rule program, includes performance monitoring of functions associated with ECCS and CSS. The inclusion of ECCS and CSS into the

Maintenance Rule program, and the assessment of acceptable system performance, provide continued assurance of the availability for performance of the required functions.

Finally, the licensee stated that condition reports would be written to document any adverse conditions identified during the containment inspections or containment emergency sumps and strainers surveillances.

The NRC staff reviewed the licensee's information and concludes that the licensee's monitoring program is acceptable because it is consistent with the guidance in RG 1.174, Section C.3.

4.0 PROGRAMMATIC ASPECTS RELIED UPON BY NRC STAFF FOR THIS REVIEW

4.1 Quality Assurance

RG 1.174,[41] Section C.5, "Quality Assurance," provides the NRC staff's position on quality assurance requirements for risk-informed changes to the licensing basis. Specifically, RG 1.174 states, in part, that:

> As stated in Section 2 of this guide [RG 1.174], the quality of the engineering analyses conducted should justify proposed LB changes will be appropriate for the nature of the change. In this regard, it is expected that for traditional engineering analyses (e.g., deterministic engineering calculations), existing provisions for quality assurance (e.g., Appendix B to 10 CFR Part 50, for safety-related SSCs) will apply and provide the appropriate quality needed. Likewise, when a risk assessment of the plant is used to provide insights into the decisionmaking process, the PRA is to have been subject to quality control.

RG 1.174 further states that when PRA information is used to enhance or modify activities affecting the safety-related functions of SSCs, four pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 should be met:

- Use personnel qualified for the analysis.
- Use procedures that ensure control of documentation, including revisions, and provide for independent review, verification, or checking of calculations and information used in the analyses. (An independent peer review or certification program can be used as an important element in this process.)
- Provide documentation and maintain records in accordance with the guidelines Section 6 of this guide [RG 1.174].
- Use procedures that ensure that appropriate attention and corrective actions are taken if assumptions, analyses, or information used in previous decision-making are changed (e.g., licensee voluntary action) or determined to be in error.

In its letters dated March 25, 2015,[19] July 21, 2016,[27] and October 20, 2016,[30] the licensee described the quality assurance program that was applied to the risk-informed assessment of debris.

The licensee stated that the risk-informed analysis was completed under the STP Operations Quality Assurance Program (OQAP) using qualified plant and contractor personnel. The OQAP requires a preparer, reviewer, and approver for calculations. This was applied to contractor personnel as well. The licensee stated that vendor-supplied documents and calculations were processed through the appropriate procedure under its quality assurance program. The licensee also employed an independent oversight group (University of Illinois at Urbana-Champaign) to review the calculations and methodologies used in the analyses. The licensee stated that its corrective action program ensures that appropriate attention and corrective actions are taken if aspects of the analysis are found to be in error.

In its October 20, 2016, letter, the licensee provided revised UFSAR pages associated with this risk-informed LAR that included a change control and reporting section in a new Appendix 6A to the STP UFSAR. The NRC staff reviewed the submitted information regarding documentation and records and concludes that the guidance set forth in Section C.6 of RG 1.174 is met because, as discussed above, the licensee has documented the analysis process under its approved Quality Assurance program.

Based on review of the information provided by the licensee, the NRC staff finds the quality assurance program used for the risk-informed approach to be acceptable because it meets the regulatory position in RG 1.174, Section C.5.

4.2 Key Elements of the Risk-Informed Analysis

In its letter dated October 20, 2016,[30] the licensee provided revised UFSAR pages for both units that identified the key elements of its risk-informed analysis and the corresponding methods, approaches, and data that would require prior NRC review and approval to change. Specifically, the licensee proposed the following UFSAR language:

> Changes to key methods and approaches of the risk-informed methodology set forth in [final NRC-approved RoverD description] are to be evaluated as a potential 'departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses' analogous to 10CFR50.59(c)(2)(viii).

The licensee stated that it cannot change these key methods and approaches under 10 CFR 50.59 without prior NRC review and approval. The UFSAR then lists the key elements of the risk-informed analysis that would be covered under the above paragraph, summarized below:

a. The methodology for quantifying the pipe break frequencies used to calculate the change in CDF and LERF.

b. The assumption that fine fiber is to be applied as the governing debris source

c. The assumptions and methods in the thermal-hydraulic analyses

d. The availability of key sources of defense in depth.

The NRC reviewed the licensee's proposed approach as described in its revised UFSAR and concludes that it is acceptable because the list of key elements is appropriate and the changes ensure that the NRC reviews and approves any changes to the key methods, approaches, and data of the risk-informed approach.

4.3 Periodic Update of the Risk-Informed Analysis

In its letter dated July 21, 2016,[27] the licensee stated it will review relevant elements of the risk-informed assessment every 48 months to ensure the continued validity of the results. Forty-eight months was chosen since it is a reasonable balance between accuracy and burden on the licensee, and consistent with common practice used in other risk-informed programs (e.g., 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors"). The NRC staff concludes that, when combined with the performance measurement strategies as described above, the update process described by the licensee provides reasonable confidence that changes to the plant that would significantly affect the risk attributable to debris would be identified and evaluated.

4.4 Reporting and Corrective Action

In its letter dated October 20, 2016,[30] the licensee provided updated pages to its UFSAR for both units that specified reporting criteria and identified its corrective action process. Specifically, the licensee proposed the following UFSAR language for reporting:

> Nonconforming conditions that make the strainer(s) inoperable for longer than required TS completion time will meet the 10CFR50.73 reporting criteria for a condition prohibited by TS. Conditions that cause the emergency sump strainers to be inoperable and result in the debris-related ΔCDF or ΔLERF to be greater than the RG 1.174 Region II acceptance guidance are to be reported in accordance with 10CFR50.72 or 10CFR50.73.

The licensee further stated that timely action will be assured by the proposed Technical Specification required action time and/or the STPNOC corrective action program.

The NRC staff finds the approach specified in the change pages to the STP UFSARs acceptable to ensure that situations in which an update to the risk-informed assessment reveals that the acceptance guidelines of RG 1.174 have been exceeded, are reported and corrected because the licensee has specified the reporting and corrective action requirements in the UFSARs.

5.0 TECHNICAL SPECIFICATIONS

5.1 Proposed Technical Specification Changes

TS 3/4.5.2

New ACTION c would state:

c. With less than the required flow paths OPERABLE solely due to potential effects of LOCA generated and transported debris that exceeds analyzed amounts, perform the following:

1. Immediately initiate action to implement compensatory actions,

AND

2. Within 90 days restore the affected flowpath(s) to OPERABLE status,

OR

Be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Current ACTION c would be renumbered as ACTION d and state:

d. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be submitted within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

TS 3/4.6.2

New ACTION c would state:

c. With one or more Containment Spray Systems inoperable in MODE 1, 2, OR 3 solely due to potential effects of LOCA generated and transported debris that exceeds analyzed amounts, perform the following:

1. immediately initiate action to implement compensatory actions,

AND

2. within 90 days restore the affected system(s) to OPERABLE status,

OR

Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

5.2 Licensee's Justification for Technical Specification Changes

The licensee proposed TS changes to create new actions with their associated completion times to recognize the impacts of debris in TS 3/4.5.2, "ECCS Subsystems – Tavg Greater Than or Equal to 350°" and TS 3/4.6.2, "Depressurization and Cooling Systems – Containment Spray System." The ECCS and CSS are the only TS SSCs that depend on the containment emergency sumps as a support system and are, therefore, the only SSCs that are directly subject to the effect of debris.

The primary function of the ECCS is to remove stored and fission product decay heat from the reactor core. In addition, the ECCS provides shutdown capability during accident conditions by means of boron injection. The ECCS is designed to tolerate a single active failure and to withstand the operating basis earthquake and the safe shutdown earthquake without loss of function. The ECCS consists of the high head safety injection and low head safety injection pumps, safety injection accumulators, residual heat removal heat exchanges, refueling water storage tank along with the associated piping, valves, instrumentation and other related equipment.

The licensee proposed changes to TS 3/4.5.2 by adding a new action 'c', renumbering existing action 'c' as 'd' and making conforming changes to the ECCS TS pages numbering to accommodate the new action statement.

The CSS is provided to reduce the concentration and quantity of fission products in the containment atmosphere and to reduce the pressure and temperature within containment following a LOCA. The CSS is designed to tolerate a single active failure and to withstand the operating basis earthquake and the safe shutdown earthquake without loss of function. The CSS also assists in reducing offsite radiological exposures resulting from a DBA to less than the limits of 10 CFR 50.67 by rapidly reducing the airborne elemental iodine and particulate concentrations in containment following a DBA.

5.3 NRC Staff Evaluation of Proposed Technical Specification Changes

The new proposed actions for both ECCS and CSS TS apply only to the potential effects of LOCA generated and transported debris that exceed the amount analyzed in the UFSAR. In other words, for those analyzed conditions where the probability of failure is very small. The new actions do not apply to conditions that have a high probability of preventing the containment emergency sumps from performing their required support function, as discussed in the questions below.

In its submittal, the licensee stated that compensatory actions required by new TS 3/4.5.2.c.1 and 3/4.6.2.c.1, would include actions such as:

- Removing the debris or source of debris, or taking action that would prevent transport of the debris to the emergency sump
- Deferring maintenance that would affect the availability of the affected systems and strainers
- Increasing the frequency of RCS leak detection monitoring
- Briefing operators on LOCA debris management actions

The licensee also stated that operability with respect to debris is based on a quantity of debris identified. Emergent nonconforming or degraded conditions affecting the quantity of analyzed debris will be evaluated using a deterministic process not a quantitative risk assessment.

The proposed completion time of "immediately" for both c.1 actions is acceptable since it places urgency on the actions that could mitigate or lessen the conditions.

In a letter dated April 11, 2016, the NRC staff requested additional information on the basis for the proposed 90-day completion time for the new c.2 actions requiring the operators to restore the affected subsystem/systems to operable status. Specifically, the NRC staff was concerned with allowing an excessive amount of time for certain scenarios. For instance, 90 days seemed to be excessive for a scenario where gross blockage of the strainer was evident and the ECCS and CSS would clearly be incapable of performing their specified safety functions (e.g., if tarps were inadvertently left covering the sump screens following an outage).

In its July 18, 2016, response, the licensee stated, in part, that:

> STPNOC did not intend the action to apply to conditions such as tarps covering the strainers. Degradation or nonconformances of this kind do not meet the proposed TS condition of "potential effects of LOCA generated and transported debris". The 90-day action would not apply and the appropriate TS action for the affected train(s) should be entered. The excerpt below from the Bases for the proposed TS changes addresses conditions such as this...
>
> > Applicability
> >
> > This required action applies only for the potential effects of debris on emergency sump strainer operability or on in-core debris effects. It does not apply for effects other than those caused by debris for which the testing and analysis apply. Debris effects are conditions caused by transportable debris that could impact the net positive suction head or otherwise degrade pump performance, or cause strainer structural failure by excess accumulation on one of more of the emergency sump strainers. Obstructions or covers on the strainers such as tarps or other conditions that are a physical degraded or nonconforming condition of the strainer (e.g., gaps, deformations) are to be addressed by the system train-specific, non-debris TS actions a and b.

The licensee also stated:

> A 90-day completion time is reasonable for emergent conditions that involve debris that could be generated and transported under LOCA conditions. The likelihood of an initiating event in the 90-day completion time is very small (1/4 of the LOCA annual frequency). There are margins in the debris generation and transport analyses and in the downstream and in-core effects analyses. Ninety days is a reasonable time to identify and implement mitigating or compensatory action, such as removing the debris, securing or containing the debris so that it is not transportable, performing additional analysis, or obtaining regulatory relief (e.g., Enforcement Discretion and/or Emergency or Exigent TS change). The

> compensatory actions required by proposed Required Action c.1 provide additional assurance that the potential debris effects of the emergent condition would be mitigated. In addition to the actions directly addressing the debris just mentioned, plant system configuration can be managed by application of the CRMP [Configuration Risk Management Program] to maximize availability of mitigating systems (e.g., ECCS, AFW [auxiliary feed water], SDGs [standby diesel generators]) and defense in depth (e.g., containment isolation, CCW [component cooling water], ECW [essential cooling water]) by limiting activities that remove them from service.

The NRC staff finds the 90-day completion time of new action c.2, proposed in TS 3/4.5.2 and 3/4.6.2, is a reasonable time to diagnose, plan and possibly lessen or mitigate the exceeded debris condition and prevent a loss of ECCS and CSS specified safety function. In addition, 90 days is reasonable given the conservatisms in the analysis and the proposed compensatory actions required to be implemented immediately. Also, TS surveillance requirements involve visual inspection of the containment and containment sumps to ensure that no loose debris is present which would be transported to the containment sumps, the subsystem suction inlets are not restricted by debris, and the sump components show no evidence of structural distress or abnormal corrosion.

If the operators are unable to restore the affected ECCS or subsystems to operable status, new action 3/4.5.2.c. requires the unit to: "Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 6 hours." This condition is acceptable since it is consistent with the existing TS and requires the operators to place the unit in a condition in which the limiting condition for operation no longer applies.

If the operators are unable to restore the affected CSS or subsystems to operable status, new action 3/4.6.2.c requires the unit to: "Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours." This condition is acceptable since it is consistent with the existing TS and requires the operators to place the unit in a condition in which the limiting condition for operation no longer applies.

The NRC staff reviewed the licensee's proposed TS changes against the regulations and concludes with reasonable assurance that the licensee will continue to meet the requirements of 10 CFR 50.36(c)(2) for the reasons discussed above. Therefore, the staff concludes that the proposed TS changes are acceptable.

6.0 RADIOLOGICAL CONSEQUENCES

By letter dated March 6, 2008,[96] the NRC staff issued amendments allowing full implementation of an alternate source term (AST) for STP. Since STP has a previously-approved AST, the relevant regulations associated with the AST and the radiological consequences of DBAs are 10 CFR 50.67, "Accident Source Term" and 10 CFR 50, Appendix A, GDC 19, "Control Room."

The NRC staff reviewed the STP GSI-191 LAR for impacts to the radiological consequences of the previously analyzed DBAs. The staff also reviewed the impacts of the LAR on compliance with 10 CFR 50.67 and GDC 19.

In addition to the amendments requested, the licensee requested exemptions from 10 CFR 50.46(a), and GDC 35, GDC 38, and GDC 41 of 10 CFR Part 50, Appendix A. The licensee proposed to comply with 10 CFR 50.46, GDC 35, GDC 38, and GDC 41 using a risk-informed approach for post-accident debris effects, based on demonstrating acceptable ECCS and CSS design to modify the licensing basis, thus, compliance with other regulatory requirements that rely on acceptable design for these systems and components would continue to be met in the current licensing basis. STPNOC requested the exemptions to use a risk-informed method to show ECCS and CSS function accounting for the effects of debris following postulated LOCAs. The risk-informed method evaluates the effects on strainer blockage and core blockage resulting from debris concerns raised by GSI-191. In order to confirm acceptable sump design, the risk associated with GSI-191 is evaluated to include the failure mechanisms associated with loss of core cooling and strainer blockage. To evaluate the accident dose consequences herein, the NRC staff assumed that the STP's proposed risk-informed alternative to resolve GSI-191 issues was found acceptable and that the requested exemptions were granted, thus confirming sump performance continued to support reliable plant design and operation (i.e., reliable ECCS and CSS performance).

In a letter dated August 20, 2015,[20] the licensee addressed the impacts to 10 CFR 50.67, "Accident Source Term," and GDC 19, "Control room," in Attachment 2-5 concluding that exemptions from those regulations were not needed.

The NRC staff reviewed the licensee's justification that exemptions from 10 CFR 50.67 and GDC 19 are not needed to use the risk-informed methodology to define the DBA for the accident-based source term. Section 50.67 identifies the major considerations for defining an accident source term for purposes of evaluating the radiological consequences of DBAs, and establishes the acceptance criteria (expressed in total effective dose equivalent) for radiological source consequence evaluations. Section 50.67 does not establish or identify the methods or approach of evaluating DBAs for determining radiological dose consequences. However, Section 50.67 states that the radiological consequences evaluation should be performed for the applicable DBAs previously analyzed in the UFSAR. Further, as provided in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000,[97] an accident source term, typically postulated to occur as a result of a large LOCA, is intended to involve significant core damage. Facility-analyzed DBAs are not intended to be actual event sequences; rather, they are intended to be surrogates to enable deterministic evaluation of the response of engineered safety features.

The risk-informed alternative used to confirm ECCS and CSS functionality does not impact the assumptions for the AST analysis. Therefore, since the risk-informed alternative to comply with ECCS and CSS functionality is only used to establish or change the progression of the DBA analyzed in the safety analysis report, and does not impact the DBA itself or its accident-based source term, then the radiological consequence analysis would not be impacted. Thus, the radiological consequence analysis does not need to be re-performed since use of the risk-informed methodology does not change the accident sequence or the source term assumed for the AST analysis, and an exemption from 10 CFR 50.67 is not needed.

The NRC staff also reviewed the licensee's justification that an exemption from GDC 19 is not needed. The NRC staff determined that GDC 19 requires, in part, design basis radiological analyses involving the radiological consequences (i.e., doses) of DBAs to ensure that nuclear power plant control rooms remain habitable. However, GDC 19 does not establish or identify the methods or approach of evaluating DBAs for determining the control room dose. The risk-informed alternative used to confirm ECCS and CSS functionality, as stated above, does not establish or change the progression of the DBA analyzed in the UFSAR, and the radiological consequence analysis is not affected. Thus, the radiological consequence analysis does not need to be re-performed since the risk-informed methodology does not change the accident sequence or the source term assumed for the AST analysis, and an exemption from GDC 19 is not needed.

Attachment 3 of the submittal dated August 20, 2015,[20] provides the proposed changes to the STP licensing basis, pursuant to NRC approval of the risk-informed approach and exemption request. The proposed exemptions do not result in any physical changes to the facility.

By e-mail dated January 14, 2014,[98] the NRC staff requested additional information from the licensee to clarify the accident dose consequence analyses by describing any changes to the current licensing basis. By letter dated March 17, 2014,[11] the licensee responded that it did not revise the licensing basis accident dose consequence analyses, including the AST analyses, for the risk-informed GSI-191 licensing application. The relevant dose results currently described in Chapter 15 of the STP UFSAR remain unchanged and continue to be valid.

The NRC staff found the licensee's response acceptable because it is not necessary to revise the licensing basis accident dose consequence analyses, including the AST analyses, if the licensee is granted exemptions. The licensee requested the exemptions to use a risk-informed method to analyze ECCS and CSS performance considering the impacts of debris. The increase in risk to ECCS and CSS operability is very small and within the Commission's Safety Goal Policy Statement. Thus, the analyses and results currently described in the STP UFSAR remain valid. This basis is clarified in the preceding discussion regarding the exemption to 10 CFR 50.67 and GDC 19.

The NRC staff also asked the licensee to clarify a condition related to the AST license amendments currently in effect at STP. Westinghouse Electric Company Nuclear Safety Advisory Letter (NSAL)-06-15, dated December 13, 2006, advised operators of Westinghouse plants that the single-failure scenario for the steam generator tube rupture (SGTR) analysis that licensees used in their accident analysis may not be limiting. As stated in the STP AST SE dated March 6, 2008:

> The licensee has evaluated the applicability of NSAL-06-15 against the accident analysis assumptions and has determined that the current single-failure assumption for the STP [steam generator tube rupture (SGTR)] analysis is not limiting. Therefore, the licensee is operating under compensatory measures to meet regulatory dose guidelines. The licensee plans to resolve this condition at the earliest opportunity so that the assumptions, including the limiting single failure, for the SGTR accident analysis described herein are consistent with the plant response to this event. To support the limiting single-failure assumptions in the SGTR analysis, STP will maintain an administrative limit for reactor coolant system dose equivalent iodine 131 so that the radiological dose reference values for the SGTR analysis remain bounding, and the licensee will continue to comply with GDC 19.

In an e-mail dated January 14, 2014,[98] the NRC staff asked the licensee if this condition had been resolved and to provide a justification that GDC 19 continues to be met. By letter dated March 17, 2014,[11] the licensee responded that the condition was resolved by a design change to make the steam valves to the moisture separator reheater fail closed. The design change has been implemented at STP. This corrective action restored the original design and licensing basis and eliminated the need for the administrative limit on dose equivalent iodine 131 that had been implemented as a compensatory action. The NRC staff finds this response acceptable because the original design basis was restored by implementation of the design change.

In an e-mail dated January 14, 2014,[98] the NRC staff asked for additional information concerning sump water pH, radiological consequences, and the loss of the CSS. By letter dated March 17, 2014,[11] the licensee responded that the sump pH history was investigated over 30 days both analytically and experimentally. The investigation showed that the STP sump pH will remain substantially above 7.0 for 30 days. Any loss of NPSH for the ECCS and CSS pumps, impeding the flow of water from the sump, has been analyzed using a risk-informed approach. The NRC staff finds the response is acceptable because the criteria of RG 1.174 is met and the sump pH was determined to remain substantially above 7.0 for 30 days in accordance with RG 1.183. The guidance from RG 1.183 specifies that the iodine deposited in the sump water can be assumed to remain in solution as long as the containment sump pH is maintained at or above 7.0.

The NRC staff reviewed the licensee's evaluation[11] against the NRC staff-accepted guidance and concludes there is reasonable assurance that the licensee will continue to meet the applicable accident dose limits following implementation of risk-informed GSI-191. The staff found that STP continues to comply with the regulatory and design basis criteria established for the AST in 10 CFR 50.67 and GDC 19. The staff also finds there is reasonable assurance that the licensee's estimates of the exclusion area boundary, low population zone, and control room doses will continue to comply with these criteria. Therefore, the staff concludes that the proposed license amendment is acceptable with respect to the radiological dose consequences of the DBAs provided the exemptions from 10 CFR 50.46, and GDC 35, 38, and 41 are granted.

7.0 TECHNICAL REVIEW CONCLUSION

The NRC staff separately evaluated the licensee's requests for 10 CFR 50.12 exemptions from specific requirements of 10 CFR 50.46, and GDCs 35, 38, and 41, to use a risk-informed methodology; however, portions of this SE provide the basis for findings necessary to grant the exemptions particularly the finding that the exemptions present no undue risk to public health and safety.[††]

The five key principles of risk-informed regulation from RG 1.174 were the focus of the NRC staff's review. The NRC staff concludes that the licensee's proposed change to its licensing basis as provided in the UFSAR is acceptable because it satisfies the key principles of risk-informed decision-making as delineated in RG 1.174, in that:

- the proposed change meets current regulations unless it is explicitly related to exemptions specifically requested and granted, as discussed in SE Section 3.1 (Key Principle 1)

- the proposed change is consistent with defense-in-depth philosophy, as discussed in SE Section 3.2 (Key Principle 2)

- the proposed change maintains sufficient safety margins, as discussed in SE Section 3.3 (Key Principle 3)

[††] The exemption requests are discussed in a separate letter (ADAMS Accession No. ML17037C871), and in a *Federal Register* notice to be published shortly.

- the increases in risk resulting from the proposed change are small and consistent with the Commission's Safety Goal Policy Statement, as discussed in SE Section 3.4 (Key Principle 4)

- the impact of the proposed change is monitored with performance measurement strategies, as discussed in SE Section 3.5 (Key Principle 5)

Additionally, the NRC staff has reviewed the proposed changes to the TSs, the UFSAR, and the impacts to the radiological consequences. As discussed in SE Section 5.0, the NRC staff finds the changes to the TSs are acceptable. As discussed in SE Section 6.0, the NRC staff finds that there are no impacts to the AST previously approved for STP. Also, the revisions to the UFSAR as set forth in Attachment 3-4 of the licensee's supplement dated August 20, 2015, and Attachment 3-4 of the licensee's supplement dated October 20, 2016, are acceptable.

Finally, as detailed in SE Attachment 2, the NRC staff concludes that the STPNOC long-term core cooling evaluation model and the simulations performed specifically for STP, Units 1 and 2, provide a conservative analysis for debris impacts on long-term core cooling for hot-leg breaks 16 inches in diameter and smaller. Further, the simulations performed with this evaluation model along with those from the LOCA Disposition Model demonstrates that the stated acceptance criteria from WCAP-16793 have been satisfied:

- The evaluation model used for the LTCC analysis demonstrates that the maximum clad temperature remains at the saturation temperature and therefore shall not exceed 800 °F following core quench and re-flooding.

- The LOCA Disposition Model analysis demonstrates that the thickness of the cladding oxide and the deposits of material on the fuel shall not exceed 0.050 inches in any fuel region.

The NRC staff's conclusions in the SE are specific to STP and the analysis performed. Future use of this in-vessel thermal-hydraulic evaluation model was not considered, because use of the evaluation model outside the simulations reviewed could invalidate the conclusions reached in this SE. If the input and modeling assumptions are made less conservative (e.g., by decreasing the severity of accident conditions, the amount of debris generated, post-LOCA PCT, or oxide thickness), or if the relative importance of specific models or flow paths are changed, then the NRC staff's assessment is no longer applicable.

8.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Texas State official was notified on June 22, 2017, of the proposed issuance of the amendment. The State official had no comments.

9.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The

Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on February 16, 2016 (81 FR 7843). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

10.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: July 11, 2017

Attachments:
1. List of References
2. In-Vessel Thermal-Hydraulic Analysis

ATTACHMENT 1

LIST OF REFERENCES

1 Rencurrel, D. W., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "South Texas Project, Units 1 and 2, Revised STP Pilot Submittal and Requests for Exemptions and License Amendment for a Risk-Informed Approach to Resolving Generic Safety Issue (GSI)-191," dated June 19, 2013 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML131750250).

2 Rencurrel, D. W., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Corrections to Information Provided in Revised STP Pilot Submittal and Requests for Exemptions and License Amendment for a Risk-Informed Approach to Resolving Generic Safety Issue (GSI)-191," dated October 3, 2013 (ADAMS Accession No. ML13295A222).

3 Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Submittal of GSI-191 Chemical Effects Test Reports," dated October 31, 2013 (ADAMS Accession No. ML13323B187).

4 Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Supplement 1 to Revised STP Pilot Submittal and Requests for Exemptions and License Amendment for a Risk-Informed Approach to Resolving Generic Safety Issue (GSI)-191," dated November 13, 2013 (ADAMS Package Accession No. ML13323A128).

5 Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Supplement 1 to Revised STP Pilot Submittal for a Risk-Informed Approach to Resolving Generic Safety Issue (GSI)-191 to Supersede and Replace the Revised Pilot Submittal," dated November 21, 2013 (ADAMS Accession No. ML13338A165).

6 Meier, M. D., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Reference Document for STP Risk-Informed GSI-191 Application," dated December 23, 2013 (ADAMS Accession No. ML14015A311).

7 Meier, M. D., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Response to STP-GSI-191-EMCB-RAI-1," dated December 23, 2013 (ADAMS Accession No. ML14015A312).

8 Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information re Use of RELAP5 in Analysis for Risk-Informed GSI-191 Licensing Application," dated January 9, 2014 (ADAMS Accession No. ML14029A533).

[9] Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Submittal of CASA Grande Code and Analyses for STP's Risk-Informed GSI-191 Licensing Application," dated February 13, 2014 (ADAMS Accession No. ML14052A053).

[10] Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Submittal of GSI-191 Chemical Effects Test Reports," dated February 27, 2014 (ADAMS Accession No. ML14072A076).

[11] Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Response to NRC Accident Dose Branch Request for Additional Information Regarding STP Risk-Informed GSI-191 Application," dated March 17, 2014 (ADAMS Accession No. ML14086A383).

[12] Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Submittal of CASA Grande Source Code for STP's Risk-Informed GSI-191 Licensing Application," dated March 18, 2014 (ADAMS Accession No. ML14087A126).

[13] Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Second Submittal of CASA Grande Source Code for STP's Risk-Informed GSI-191 Licensing Application," dated May 15, 2014 (ADAMS Accession No. ML14149A354).

[14] Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Revised Affidavit for Withholding for CASA Grande Analysis for STP's Risk-Informed GSI-191 Licensing Application," dated May 15, 2014 (ADAMS Accession No. ML14149A353).

[15] Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "First Set of Responses to April, 2014, Requests for Additional Information Regarding STP Risk-Informed GSI-191 Licensing Application Revised," dated May 22, 2014 (ADAMS Accession No. ML14149A434).

[16] Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Second Set of Responses to April, 2014, Requests for Additional Information Regarding STP Risk-Informed GSI-191 Licensing Application," dated June 25, 2014 (ADAMS Accession No. ML14178A481).

[17] Powell, G. T, STP Nuclear Operating Company, "Third Set of Responses to April, 2014, Requests for Additional Information Regarding STP Risk-Informed GSI-191 Licensing Application," dated July 15, 2014 (ADAMS Accession No. ML14202A045).

[18] Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Submittal of Updated CASA Grande Input for STP's Risk-Informed GSI-191 Licensing Application," dated March 10, 2015 (ADAMS Accession No. ML15072A092).

19 Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Description of Revised Risk-Informed Methodology and Responses to Round 2 Requests for Additional Information Regarding STP Risk-Informed GSI-191 Licensing Application," dated March 25, 2015 (ADAMS Accession No. ML15091A440).

20 Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Supplement 2 to STP Pilot Submittal and Requests for Exemptions and License Amendment for a Risk-Informed Approach to Address Generic Safety Issue (GSI)-191 and Respond to Generic Letter (GL) 2004-02," dated August 20, 2015 (ADAMS Package Accession No. ML15246A125).

21 Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Revision to Proposed Exemption to 10CFR50.46 Described in STP Pilot Submittal and Requests for Exemptions and License Amendment for a Risk-Informed Approach to Address Generic Safety Issue (GSI)-191 and Respond to Generic Letter (GL) 2004-02," dated April 13, 2016 (ADAMS Accession No. ML16111B204).

22 Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "First Set of Responses to April 11, 2016 Requests for Additional Information Regarding STP Risk-Informed GSI-191 Licensing Application," dated May 11, 2016 (ADAMS Accession No. ML16154A117).

23 Connolly, J., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Applicability of Application Supplement 1 Correspondence to Supplement 2 to STP Risk-Informed GSI-191 Licensing Application," dated June 9, 2016 (ADAMS Accession No. ML16176A148).

24 Connolly, J., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Second Set of Responses to April 11, 2016 Requests for Additional Information Regarding Risk-Informed GSI-191 Licensing Application," dated June 16, 2016 (ADAMS Accession No. ML16196A241).

25 Connolly, J., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Third Set of Responses to April 11, 2016 Requests for Additional Information Regarding STP Risk-Informed GSI-191 Licensing Application," dated July 18, 2016 (ADAMS Accession No. ML16209A226).

26 Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "APLA Round 4 Requests for Additional Information Regarding STP Risk-Informed GSI-191 Licensing Application," dated July 21, 2016 (ADAMS Accession No. ML16230A232).

27 Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Third Set of Responses to April 11, 2016 Requests for Additional Information Regarding STP Risk-Informed GSI-191 Licensing Application, Response to SNPB RAIs," dated July 21, 2016 (ADAMS Accession No. ML16229A189).

28 Connolly, J., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Response to RAI 4-2 of APLA Round 4 Requests for Additional Information Regarding STP Risk-Informed GSI-191 Licensing Application," dated July 28, 2016 (ADAMS Accession No. ML16221A393).

[29] Connolly, J., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Revised Response to Request for Additional Information APLA-4-4 Re: STP Risk-Informed GSI-191 Licensing Application," dated September 12, 2016 (ADAMS Accession No. ML16272A162).

[30] Connolly, J., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Supplement 3 to Revised Pilot Submittal and Requests for Exemptions and License Amendment for a Risk-Informed Approach to Address Generic Safety Issue (GSI)-191 and Respond to Generic Letter (GL) 2004-02," dated October 20, 2016 (ADAMS Accession No. ML16302A015).

[31] Connolly, J., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Additional Information Regarding Sensitivity Studies for STPNOC Risk-Informed Pilot GSI-191 Application," dated November 9, 2016 (ADAMS Accession No. ML16321A407).

[32] Rencurrel, D. W., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Revised Applicability Matrix for Response to Request for Additional Information Questions APLA-1a and APLA-1b Regarding STP Risk-Informed GSI-191 Licensing Application," dated December 7, 2016 (ADAMS Accession No. ML16365A006).

[33] Connolly, J., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "South Texas Project Unit 1 and 2 - Response to Request for Additional Information on Revised Applicability Matrix for Questions Regarding Risk-Informed GSI-191 Licensing Application (TAC Nos MF2400 and MF2401)," dated January 19, 2017 (ADAMS Accession No. ML17025A123).

[34] U.S. Nuclear Regulatory Commission, Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004 (ADAMS Accession No. ML042360586).

[35] U.S. Nuclear Regulatory Commission, Staff Requirements Memorandum SRM-SECY-12-0093, "Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," dated December 14, 2012 (ADAMS Accession No. ML12349A378).

[36] U.S. Nuclear Regulatory Commission, NUREG-0897, Revision 1, "Containment Emergency Sump Performance, Technical Findings Related to Unresolved Safety Issue A-43," October 1985 (ADAMS Accession No. ML112440046).

[37] U.S. Nuclear Regulatory Commission, Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," dated June 9, 2003 (ADAMS Accession No. ML031600259).

[38] U.S. Nuclear Regulatory Commission, Staff Requirements Memorandum SRM-SECY-10-113, "Closure Options for Generic Safety Issue-191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance," dated December 23, 2010 (ADAMS Accession No. ML103570354).

[39] U.S. Nuclear Regulatory Commission, SECY-12-0093, "Closure Options for Generic Safety Issue - 191, Assessment of Debris Accumulation on Pressurized-Water Reactor Sump Performance," dated June 9, 2012 (ADAMS Accession No. ML121310648).

[40] Rencurrel, D. W., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "South Texas Project, Units 1 and 2 – GSI 191 Resolution Path Schedule and Commitment Changes," dated June 4, 2012 (ADAMS Accession No. ML13025A360).

[41] U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," May 2011 (ADAMS Accession No. ML100910006).

[42] U.S. Nuclear Regulatory Commission, "Safety Goals for the Operations of Nuclear Power Plants: Policy Statement," *Federal Register*, Volume 51, p. 30028, Washington, DC, August 4, 1986.

[43] U.S. Nuclear Regulatory Commission, "Use of Probabilistic Risk Assessment Methods in Nuclear Activities; Final Policy Statement," *Federal Register*, Volume 60, Number 158, p. 42622, Washington, DC, August 16, 1995.

[44] PWR Owners Group, "Transmittal of WCAP-16793-NP-A, Revision 2, 'Evaluation of Long Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid,' dated July 2013," dated August 13, 2013 (ADAMS Package Accession No. ML13239A111).

[45] U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 15.0.2, "Review of Transient and Accident Analysis Methods," December 2005 (ADAMS Accession No. ML053550265).

[46] Nuclear Energy Institute, "Transmittal of 'PWR Containment Sump Evaluation Methodology,'" dated May 28, 2004 (ADAMS Package Accession No. ML041550661).

[47] Black, S., U.S. Nuclear Regulatory Commission, letter with safety evaluation to Anthony R. Pietrangelo, Nuclear Energy Institute, "Pressurized Water Reactor Containment Sump Evaluation Methodology," dated December 6, 2004 (ADAMS Package Accession No. ML043280641).

[48] Nuclear Energy Institute, NEI 04-07, Volume 1, "Pressurized Water Reactor Sump Performance Evaluation Methodology," December 2004 (ADAMS Accession No. ML050550138); Volume 2, "Safety Evaluation by the Office of Nuclear Reactor Regulation re to NRC Generic Letter 2004-02, Revision 0," dated December 6, 2004 (ADAMS Accession No. ML050550156).

[49] Westinghouse Electric Company, LLC, WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," March 2008 (ADAMS Accession No. ML081150379).

50 Westinghouse Electric Company, LLC, WCAP-16406-P-A, Revision 1, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," March 2008 (Not publicly available. Proprietary information).

51 U.S. Nuclear Regulatory Commission, "Revised Content Guide for Generic Letter 2004-02 Supplemental Responses," dated November 21, 2007 (ADAMS Package Accession No. ML073110389).

52 U.S. Nuclear Regulatory Commission, "Revised Review Guidance for Licensee Responses to Generic Letter 2004-02," dated March 28, 2008 (ADAMS Package Accession No. ML080230234).

53 U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments," September 2013 (ADAMS Accession No. ML13198A250).

54 U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007 (ADAMS Accession No. ML071700658).

55 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009 (ADAMS Accession No. ML090410014).

56 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-Of-Coolant Accident," March 2012 (ADAMS Accession No. ML111330278).

57 U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," September 2012 (ADAMS Accession No. ML12193A107).

58 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.45, Revision 1, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," May 2008 (ADAMS Accession No. ML073200271).

59 Nuclear Energy Institute, NEI 03-08, Revision 2, "Guidelines for the Management of Materials Issues," January 2010 (ADAMS Accession No. ML101050337).

60 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.14, Revision 1, "Reactor Coolant Pump Flywheel Integrity," August 1975 (ADAMS Accession No. ML003739936).

61 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.200 Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," January 2007 (ADAMS Accession No. ML070240001).

62 U.S. Nuclear Regulatory Commission, NUREG-1903, "Seismic Considerations for the Transition Break Size," February 2008 (ADAMS Accession No. ML080880140).

63 U.S. Nuclear Regulatory Commission, NUREG-1829, Volume 1, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process, Main Report," April 2008 (ADAMS Accession No. ML082250436); Volume 2, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process, Appendices A through M," April 2008 (ADAMS Accession No. ML081060300).

64 American Society of Mechanical Engineers / American Nuclear Society, ASME/ANS RA-Sa 2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers, New York, NY, and American Nuclear Society, La Grange Park, IL, 2009.

65 Boska, J. P., U.S. Nuclear Regulatory Commission, letter to Entergy Nuclear Operations, Inc., "Indian Point Nuclear Generating Unit Nos. 2 and 3 – Report on Results of Staff Audit of Corrective Actions to Address Generic Letter 2004-02 (TAC Nos. MC4689 and MC4690)," dated July 29, 2008 (ADAMS Package Accession No. ML082050446).

66 Thadani, M. C., U.S. Nuclear Regulatory Commission, letter to Edward D. Halpin, STP Nuclear Operating Company, "South Texas Project, Units 1 and 2 - RE: Request For Additional Information for Generic Letter 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors' (TAC Nos. MC4719 and MC4720)," dated December 23, 2009 (ADAMS Accession No. ML093410607).

67 Regner, L. M., U.S. Nuclear Regulatory Commission, letter to Dennis L. Koehl, STP Nuclear Operating Company, "South Texas Project, Units 1 and 2 - Request for Additional Information, Related to Request for Exemptions and License Amendment Request for Use of a Risk-Informed Approach to Resolve the Issue of Potential Impact of Debris Blockage on Emergency Recirculation during Design-Basis Accidents at Pressurized-Water Reactors (CAC Nos. MF2400-MF2409)," dated April 11, 2016 (ADAMS Accession No. ML16082A507).

68 Regner, L. M., U.S. Nuclear Regulatory Commission, letter to Dennis L. Koehl, "South Texas Project, Units 1 and 2 - Staff Audit Summary Related to Request for Exemptions and License Amendment for Use of a Risk-Informed Approach to Resolve the Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized Water Reactors (TAC Nos. MF2400, MF2401, MF2402, MF2403, MF2404, MF2405, MF2406, MF2407, MF2408, and MF2409)," dated July 29, 2015 (ADAMS Accession No. ML15175A024).

69 Nuclear Energy Institute, NEI 02-01, Revision 1, "Condition Assessment Guidelines: Debris Sources Inside PWR Containments," September 2002 (ADAMS Accession No. ML030420318).

[70] Westinghouse Electric Company, LLC, WCAP-16568-P, "Jet Impingement Testing to Determine the Zone of Influence (ZOI) for DBA-Qualified/Acceptable Coatings,'" June 2006 (Not publicly available. Proprietary information).

[71] Ruland, W. H., U.S. Nuclear Regulatory Commission, letter to Alexander Marion, Nuclear Energy Institute, "Revised Guidance Regarding Coatings Zone of Influence for Review of Final Licensee Responses to Generic Letter 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors,'" dated April 6, 2010 (ADAMS Accession No. ML100960495).

[72] Blevins, M., TXU Generation Company, LP, Letter to U.S. Nuclear Regulatory Commission, "Comanche Peak Steam Electric Station (CPSES), Docket Nos. 50-445 And 50-446, 'Transmittal of Report on TXU Power Sponsored Coatings Performance Test "Comanche Peak/TXU/Keeler & Long Report on Coating Performance Test," dated October 20, 2006 (ADAMS Package Accession No. ML070230384).

[73] Choromokos, R., Alion Science and Technology, letter to U.S. Nuclear Regulatory Commission, "ALION Science and Technology – Proprietary Documents GSI-191 Low Density Fiberglass Erosion Testing," dated April 8, 2010; includes ALION-REP-ALION-1006-04, "Erosion Testing of Small Pieces of Low Density Fiberglass Debris – Test Report," March 2010 (Not publicly available. Proprietary information).

[74] Ruland, W. H., U.S. Nuclear Regulatory Commission, letter to Robert Choromokos, Alion Science and Technology, "Proprietary Erosion Testing of Submerged Nukon Low-density Fiberglass Insulation in Support of Generic Safety Issue 191 Strainer Performance Analyses," dated June 30, 2010 (ADAMS Accession No. ML101540221).

[75] Rencurrel, D. W., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "South Texas Project Units 1 and 2, Supplement 4 to the Response to Generic Letter 2004-02 (TAC Nos. MC4719 & MC4720)," dated December 11, 2008 (ADAMS Accession No. ML083520326).

[76] American Institute of Steel Construction, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, 7th Edition" (AISC 7th Edition).

[77] American National Standards Institute/American Institute of Steel Construction (AISC) N690-1994, "Specification for the Design, Fabrication, and Erection of Steel Safety Related Structure for Nuclear Facilities."

[78] Singal, B., U.S. Nuclear Regulatory Commission, letter to Dennis Koehl, STP Nuclear Operating Company, "South Texas Project, Units 1 and 2 – Request for Additional Information Related to Request for Exemptions and License Amendment Request for Use of a Risk-Informed Approach to Resolve the Issue of Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors," dated April 15, 2014 (ADAM Accession No. ML14087A075).

79 Connolly, J., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, Response to ACRS Subcommittee Question re Containment Spray Flow Considered for Single Train Case in Supplement 3 to STP Risk-Informed GSI-191 Licensing Application (TAC Nos. MF2400 and MF2401)," dated April 20, 2017 (ADAMS Accession No. ML17110A387).

80 Westinghouse Electric Company, LLC, "WCAP-10325-P-A Methodology Material Properties as Applied in the Watts Bar Containment Analysis (Non-Proprietary)," dated July 23, 2015 (ADAMS Accession No. ML15209A892).

81 Westinghouse Electric Company, LLC, Nuclear Safety Advisory Letter (NSAL) 06-6, "LOCA Mass and Energy Release Analysis," dated June 6, 2006.

82 Westinghouse Electric Company, LLC, Nuclear Safety Advisory Letter (NSAL) 11-5, "Westinghouse LOCA Mass and Energy Release Calculation Issues," dated July 25, 2011 (ADAMS Accession No. ML13239A479).

83 Westinghouse Electric Company, LLC, Nuclear Safety Advisory Letter (NSAL) 14-2, "Westinghouse Loss-of-Coolant Accident Mass and Energy Release Calculation Issue for Steam Generator Tube Material Properties," dated March 31, 2014.

84 Westinghouse Electric Company, LLC, Westinghouse InfoGram IG-14-1, "Material Properties for Loss-of-Coolant Accident Mass and Energy Release Analyses," November 5, 2014.

85 Nemeth, P., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Request for Additional Information Regarding the License Amendment Request for Extension of Containment Leakage Rate Testing Program," dated March 17, 2016 (ADAMS Accession No. ML16089A406).

86 Westinghouse Electric Company, LLC, WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version," May 1983 (Not publicly available. Proprietary information).

87 Bischoff, G., Dominion Resources Services, Inc., letter to U.S. Nuclear Regulatory Commission, "Approved Topical Report DOM-NAF-3 NP-A, 'GOTHIC Methodology for Analyzing the Response to Postulated Pipe Ruptures Inside Containment,'" dated November 6, 2006 (ADAMS Accession No. ML063190467).

88 Regner, L. M., U.S. Nuclear Regulatory Commission, letter to Dennis Koehl, STP Nuclear Operating Company, "South Texas Project, Units 1 and 2 - Issuance of Amendments Re: Request to Extend the 10-Year Containment Integrated Leak Rate Test Frequency to 15 Years (CAC Nos. MF6176 and MF6177)," dated April 29, 2016, (ADAMS Accession No. ML16116A007).

89 Singal, B. K., U.S. Nuclear Regulatory Commission, letter to Dennis Koehl, STP Nuclear Operating Company, "South Texas Project, Units 1 and 2 - Staff Audit Report Related to Request for Exemptions and License Amendment for Use of a Risk-Informed Approach to Resolve the Issue of Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors," dated January 15, 2015 (ADAMS Accession No. ML14321A677).

90 Regner, L. M., U.S. Nuclear Regulatory Commission, letter to Dennis Koehl, STP Nuclear Operating Company, "South Texas Project, Units 1 and 2 – Request for Additional Information, Round 2, Request for Exemptions and Revised Pilot License Amendment Request for a Risk-Informed Approach to Resolve Generic Safety Issue (GSI)-191 (TAC Nos. MF2400 through MF2409)," dated March 3, 2015 (ADAMS Accession No. ML14357A171).

91 U.S. Nuclear Regulatory Commission, "Trip Report on Staff Observation of Alden Research Laboratory Inc. Testing of Performance Contracting, Inc. PWR Suction Strainer," dated August 11, 2005 (ADAMS Accession No. ML052060337); "Trip Report Regarding Staff Observations of Scaled Flume Testing of the Point Beach Nuclear Plant Proposed Replacement Suction Strainer Design," dated April 20, 2006 (ADAMS Accession No. ML060750340); "Memo with Enclosures 1 and 2 - Trip Report Regarding Staff Observations Regarding Flume Testing of a Prototype Portion of the Proposed Replacement Suction Screen Design for the Comanche Peak Steam Electric Station," dated June 30, 2006 (ADAMS Accession No. ML061280580); "Staff Observations From January 2008 Trip to the PCI/Alden Test Facility to Observe head Loss Testing For Wolf Creek and Callaway Plants," dated June 24, 2008 (ADAMS Accession No. ML081830645); "Staff Observations of Testing for Generic Safety Issue 191 During February 12 and February 13 Trip to the Alden Test Facility for PCI Strainer Tests," dated April 30, 2008 (ADAMS Accession No. ML080920398); and "Saint Lucie, Unit 1 - Staff Observations of Sump Strainer Head Loss Testing at Alden Laboratory for Generic Safety Issue 191," dated September 8, 2015 (ADAMS Accession No. ML15240A154).

92 U.S. Nuclear Regulatory Commission, "Staff Observations of Testing for Generic Issue 191 During a July 29 through July 31, 2008, Trip to the Alden Test Facility for PCI Strainer Tests," dated December 16, 2008 (ADAMS Accession No. ML083470317).

93 U.S. Nuclear Regulatory Commission, "Final Safety Evaluation for Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR) WCAP-16530-NP, 'Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191,'" dated December 21, 2007 (ADAMS Accession No. ML073520891).

94 Nieh, H. K., U.S. Nuclear Regulatory Commission, letter to Gordon Bischoff, Westinghouse Electric Company, LLC, "Final Safety Evaluation for Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR) WCAP-16406-P, 'Evaluation of Downstream Sump Debris Effects in Support of GSI-191," Revision 1," dated December 20, 2007 (ADAMS Accession Nos. ML073480324 and ML073520295).

95 NUREG-1855, Vol. 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," March 2009 (ADAMS Accession No. ML090970525).

96 Thadani, M. C., U.S. Nuclear Regulatory Commission, letter to James J. Sheppard, STP Nuclear Operating Company, "South Texas Project, Units 1 and 2 - Issuance of Amendments Re: Adoption of Alternate Radiological Source Term in Assessment of Design-Basis Accident Dose Consequences," dated March 6, 2008 (ADAMS Package Accession No. ML080300062).

97 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792).

98 George, A., U.S. Nuclear Regulatory Commission, e-mail to A. Harrison, STP Nuclear Operating Company, "2014/01/14 NRR E-mail Capture - RAI's Regarding South Texas Project's Submittal for Risk-Informed Resolution of GSI-191 (TAC Nos. MF2400 and MF2401)," dated January 14, 2014 (ADAMS Accession No. ML14015A045).

ATTACHMENT 2

IN-VESSEL THERMAL-HYDRAULIC ANALYSIS ASSOCIATED WITH

LICENSE AMENDMENT FOR A RISK-INFORMED APPROACH TO ADDRESS

GENERIC SAFETY ISSUE 191 AND GENERIC LETTER 2004-02

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

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A.1 <u>INTRODUCTION</u>

In September 2004, the U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors" (Reference 1), as a result of the NRC evaluation of Generic Safety Issue 191 (GSI-191), "Assessment of Debris Accumulation on PWR [Pressurized-Water Reactor] Sump Performance." The GL 2004-02 requested that licensees for PWRs perform evaluations of the emergency core cooling system (ECCS) and the containment spray system (CSS) to assess the potential for debris entrained in the circulated containment pool to block the ECCS recirculation flow path and within the reactor and fuel assemblies following a loss-of-coolant accident (LOCA).

In December 2004, the Nuclear Energy Institute (NEI) published NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology" (Reference 2), providing a method for licensees to resolve some aspects of the concerns discussed in GL 2004-02. The NRC staff's safety evaluation (SE) of NEI 04-07 (Reference 3) found that additional guidance was needed in the area of blockage in the reactor vessel in order to adequately address the downstream effects of debris that passes through the ECCS sump strainer(s).

In response to the NRC SE's conclusions on NEI 04-07, the Pressurized Water Reactor Owners Group (PWROG) sponsored development of Topical Report (TR) WCAP-16793-NP-A, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid" (WCAP-16793) (Reference 4). WCAP-16793 provided a methodology to evaluate the effects of debris and chemical precipitates on core cooling when the ECCS is aligned to the containment sump. The objective of WCAP-16793 was to provide guidance on demonstrating that long-term core cooling (LTCC) would be maintained following a LOCA to satisfy the requirements of Title 10 of the *Code of Federal Regulations* Section 50.46 (10 CFR 50.46). WCAP-16793, Revision 2 (Reference 4), was approved with limitations and restrictions specified in the incorporated NRC SE (Reference 5).

On November 21, 2007, the NRC issued "Revised Content Guide for Generic Letter 2004-02 Supplemental Responses" (References 6 and 7), for guidance to licensees preparing supplemental responses to GL 2004-02. The Revised Content Guide provided information to licensees on how to evaluate the effects of debris carried downstream of the containment sump screen and into the reactor vessel, and show that the in-vessel effects evaluation is consistent with, or bounded by, the industry generic guidance (WCAP-16793), as modified by NRC staff comments on that document. The Revised Content Guide also identified NRC staff information needs on the application of the methods, the exceptions to WCAP-16793, and the summary of the evaluation of those areas.

By letter dated March 8, 2005 (Reference 8), the STP Nuclear Operating Company (STPNOC) submitted its 90-day response to GL 2004-02 for the South Texas Project, Units 1 and 2 (STP).

Table 1[1] contains the key correspondence which specifically addresses the in-vessel thermal-hydraulic effects of debris on LTCC for STP. This table is not a complete list of material submitted on the in-vessel thermal-hydraulic effects review, but it is a list of documents relevant to the current approach.

Table 1: List of Correspondence Related to Thermal-Hydraulic Analysis

NRC	GL 2004-02	September 13, 2004	1
NEI	PWR Sump Performance Evaluation Model (EM)	December 2004	2
NEI	SE for PWR Sump Performance EM	December 2004	3
PWROG	WCAP-16793-NP, Revision 2	July 2013	4
NRC	Revised Guidance Letter	November 21, 2007	7
NRC	Revised Guidance on GL 2004-02	November 2007	6
STPNOC	90-Day Response to GL 2004-02	March 8, 2005	8
INL	RELAP5-3D Manuals	July 2014	9
STPNOC	Reactor Coolant System (RCS) Thermal Hydraulics (TH)	August 20, 2015	10
NRC	Draft Quality Assurance (QA) Request for Additional Information (RAI)	October 21, 2015	11
NRC	Draft TH RAIs	December 11, 2015	12
NRC	RAI – Round 3	April 11, 2016	13
NRC	Methodology Audit Report	April 13, 2016	14
NRC	QA Audit Report	May 11, 2016	15
STPNOC	Part 1 of Response to RAI Round 3	May 11, 2016	16
STPNOC	Part 2 of Response to RAI Round 3	June 16, 2016	17
STPNOC	Part 3 of Response to RAI Round 3	July 21, 2016	18
STPNOC	RAI Round 3 Response Supplement	October 20, 2016	19
STPNOC	RAI Round 3 Response Supplement	November 9, 2016	20

The NRC staff issued an RAI specifically on the issue of in-vessel thermal hydraulic effects. This was considered RAI, Round 3; however, due to the STPNOC methodology change, previous in-vessel thermal-hydraulic RAI questions were superseded. See the NRC staff's letter dated December 12, 2016, regarding closeout of RAI questions no longer applicable to the GSI-191 review (Reference 21).

[1] For clarity, correspondence from the NRC is highlighted in gray. Due to the nature of this pilot program, much of the documentation submitted was changed due to a change in the approach by STPNOC.

General information for each RAI question is given in Table 2 including the question number, topic, associated SE section, and the reference number(s) of its response.

Table 2: List of RAIs

Question	Subject	Section	Reference of Response
SNPB-3-1	Clad oxide	A.4	16, 19
SNPB-3-2	Accident scenario progression	A.4.2.1.2	18, 19
SNPB-3-3	Core bypass flow	A.4.2.1.4	16
SNPB-3-4	Important phenomena	A.4.2.1.3	16
SNPB-3-5	Debris at grid spacers	A.4.2.1.3	16, 19
SNPB-3-6	Initial and boundary conditions	A.4.2.1.4	18
SNPB-3-7	Initial and boundary conditions for long-term	A.4.2.1.4	18, 19
SNPB-3-8	Phenomena modeled	A.4.2.2.2	16
SNPB-3-9	Reference and limits of closure relationships	A.4.2.2.3	17
SNPB-3-10	User manual	A.4.2.2.4	17
SNPB-3-11	Modeling of important phenomena	A.4.2.3.2	16
SNPB-3-12	Field equations	A.4.2.3.3	16
SNPB-3-13	Validation of closure relationships	A.4.2.3.4	16, 19
SNPB-3-14	Simplifications and averaging	A.4.2.3.5	16
SNPB-3-15	Level of detail	A.4.2.3.6	18
SNPB-3-16	Single version of the EM	A.4.2.4.1	16
SNPB-3-17	Validation of the EM	A.4.2.4.2	18, 19
SNPB-3-18	Mesh size sensitivity	A.4.2.4.7	18, 19
SNPB-3-19	Initial test cases	A.4.2.6.1	16, 19
SNPB-3-20	Specific sensitivity studies	A.4.2.4.7	18, 19, 20
SNPB-3-21	Important sources of uncertainty	A.4.2.5.1	18
SNPB-3-22	Uncertainties and design margin	A.4.2.5.1	18, 19
SNPB-3-23	Quality assurance program for the EM	A.4.2.6.1	18
SNPB-3-24	Input verification	A.4.2.6.1	18
SNPB-3-25	Proper convergence	A.4.2.6.1	18
SNPB-3-26	Non-physical results	A.4.2.6.1	18
SNPB-3-27	Realistic results	A.4.2.6.1	18
SNPB-3-28	Boundary conditions as prescribed	A.4.2.6.1	18
SNPB-3-29	Thoroughly understood results	A.4.2.6.1	18
SNPB-3-30	Quality assurance program documentation	A.4.2.6.2	18
SNPB-3-31	Independent peer review	A.4.2.6.2	18
SNPB-3-32	Important sources of uncertainty	A.4.2.5.1	18

A.2 REGULATORY EVALUATION

Generic Letter (GL) 2004-02 requested that holders of operating licenses for PWRs perform evaluations of the ECCS and the CSS recirculation functions considering the effects of debris following a LOCA. These evaluations are to include the potential for debris blockage at flow restrictions within the ECCS recirculation flow path downstream of the sump strainer, including potential blockage at fuel assembly inlet debris strainers and other potential flow restrictions, such as the fuel assembly spacer grids. Debris blockage at these locations has the potential to impede or prevent the flow of coolant to the reactor core, potentially leading to inadequate LTCC.

The acceptance criteria for the performance of a nuclear reactor core following a LOCA are found in 10 CFR 50.46. The regulations under 10 CFR 50.46(a)(1)(i) state that the ECCS cooling performance must be calculated in accordance with an acceptable EM. This EM is defined in 10 CFR Subsection 50.46(c)(2):

> An *evaluation model* is defined as the calculational framework for evaluating behavior of a system of the reactor system during a postulated loss-of-coolant accident (LOCA). It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

The EM must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a LOCA, make comparisons to applicable experimental data, and must identify, quantify, and assess uncertainties in the analysis method and inputs.

The acceptance criterion dealing with the long-term cooling phase of the accident recovery is in 10 CFR 50.46(b)(5), which reads as follows:

> *Long-term cooling:* After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

As stated in Section A.1, "Introduction," of this In-Vessel Thermal-Hydraulic Analysis, the NRC staff endorsed WCAP-16793 as an acceptable evaluation method to show compliance with the long-term core cooling requirements in 10 CFR 50.46 considering the effects of debris in the ECCS recirculating fluid. In the SE for WCAP-16793, the NRC staff specified that meeting 10 CFR 50.46(b)(5) requires: (1) acceptance criteria for LTCC once the core has quenched and reflooded, and (2) the mission time that should be used in evaluating debris ingestion effects on the reactor fuel.

To summarize, long-term cooling capability must be provided despite potential challenges from chemical effects (e.g., boron precipitation,[2] interaction of debris with chemicals from coatings) or physical effects (e.g., debris), as demonstrated by no significant increase in calculated peak cladding temperature (PCT). After quench and reflood, moderate increases in cladding temperature, on the order of 200 to 400 degrees Fahrenheit (°F) could be acceptable, if appropriately justified. In addition, adequate core cooling performance during the ECCS mission time is demonstrated when bulk and local temperatures are shown to be stable or continuously decreasing with the additional assurance that any debris entrained in the cooling water supply would not be capable of affecting the stable heat removal mechanism due to sump strainer clogging or downstream effects.

In WCAP-16793, the acceptance criteria for LTCC following core quench and reflooding are given as the following:

1. The maximum clad temperature shall not exceed 800 °F following core quench and reflooding

2. The thickness of the cladding oxide and the deposits of material on the fuel shall not exceed 0.050 inches in any fuel region.

The acceptance criteria do not represent, nor are they intended to be, new or additional LTCC requirements beyond the requirements in 10 CFR 50.46. Instead, they allow demonstration that local temperatures in the core are stable or continuously decreasing and that debris entrained in the cooling water supply will not affect decay heat removal. The 800 °F temperature was determined based on autoclave data that demonstrated oxidation and hydrogen pickup to be acceptable at and below 800 °F. A discussion of the technical basis for the 800 °F temperature is given in Appendix A of WCAP-16793. The 0.050-inch limit for oxide plus deposits was selected to preclude the formation of deposits that would bridge the space between adjacent rods and block flow between fuel channels.

The licensee performed a number of simulations to demonstrate that these criteria have been met, and the NRC staff reviewed these simulations. To assure the quality and uniformity of NRC staff reviews, the NRC created NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP) (Reference 22), to guide the staff in performing its reviews. Regulatory guidance for the review of design basis accident evaluation methodologies is provided in Section 15.0.2 of the SRP, "Review of Transient and Accident Analysis Methods" (Reference 23). Similar guidance is also set forth for the industry in Regulatory Guide 1.203, "Transient and Accident Analysis Methods," December 2005 (Reference 24).

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[2] Section 8 of WCAP-16793 states that the effects of boron precipitation on LTCC are being addressed by the PWROG in a separate program. Refer to SE Section 3.4.2.8, "Impact of Debris," for a description of the program and the NRC staff evaluation.

A.3 TECHNICAL EVALUATION OVERVIEW

A.3.1 NRC Staff Evaluation Method

In order to demonstrate that the impacts of in-vessel debris have been appropriately captured, STPNOC stated it performed analyses confirming that the following acceptance criteria, set forth during the NRC's review of WCAP-16793, have been satisfied:

1. The maximum clad temperature shall not exceed 800 °F following core quench and reflooding

2. The thickness of the cladding oxide and the deposits of material on the fuel shall not exceed 0.050 inches in any fuel region.

The purpose of the NRC staff's review was to determine if there is adequate core cooling during the long-term period following a LOCA (i.e., after reflood and core quench) such that the maximum clad temperature would not exceed 800 °F and the cladding oxide would not exceed 0.050 inches.

The licensee's submittal proposed to demonstrate that both of these acceptance criteria were satisfied using scientific computer simulations of the LTCC phase of the LOCA. STPNOC's computer simulations for the LTCC phase of the accident used the RELAP5-3D platform, an EM that has not previously been reviewed and approved by the NRC staff for use in this manner. Thus, the NRC staff's technical evaluation focused on determining whether the EM, when used in the manner prescribed by STPNOC, resulted in appropriate simulations of the given scenario such that the NRC staff would have confidence in the outcomes. Then, the outcomes were compared against the acceptance criteria in WCAP-16793.

It is important to note that the NRC staff's review of the EM was "simulation" focused and not "code" focused. Thus, in its scientific computer simulation review,[3] the NRC staff was concerned with determining if the simulation results were trustworthy and how trustworthy they needed to be for the intended purpose.

This distinction is important, as the guidance typically used by the NRC staff for reviews like this (e.g., SRP Section 15.0.2, Regulatory Guide 1.203) focuses on reviewing the EM. When such reviews are performed, a plant-specific simulation is not usually reviewed since the goal is to evaluate the model for a variety of future uses by the nuclear industry.

Thus, the NRC staff did not review the STPNOC LTCC EM for general use. The NRC staff's review focused on those simulations performed by STPNOC to demonstrate adequate LTCC capability in the presence of debris for STP. In that context, the NRC staff reviewed the simulations produced by the STP LTCC EM and determined whether there is reasonable assurance that those simulations are adequate representations of the STP LTCC scenario.

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[3] The topic of scientific computer simulation review is discussed in more detail in Kaizer, Heller, and Oberkampf (Reference 25).

A.3.2 Scope of the Review

The licensee deterministically modeled the phenomena during the long-term phase following small and medium hot-leg breaks using RELAP5-3D. RELAP5-3D has never been submitted for NRC review and approval, and this review is not intended to provide a generic review. The NRC staff restricted its review to focus on whether RELAP5-3D is an acceptable EM to predict the phenomena occurring during the long-term cooling phase following a LOCA at STP. This review focused on ensuring that the acceptance criteria were met for STP's current licensed operating conditions. Use of RELAP5-3D by other plants, for other purposes, or with different key inputs would require prior review and approval by the NRC.

RELAP5-3D was used by the licensee to predict the blowdown, refill, and reflood phases of the LOCA; however, the NRC staff reviewed those phases only to determine if they would provide a reasonable estimate of the initial condition for the long-term phase and only for the specific scenarios considered by STPNOC. Thus, the ability of RELAP5-3D to accurately simulate phenomena impacting the figures of merit[4] during blowdown, refill, and reflood is beyond the scope of this review for those phenomena which are not important for calculating the initial conditions of the long-term phase. Additionally, the ability of the EM to accurately simulate this scenario for other inputs is beyond the scope of this review.

As stated above, the review for in-vessel thermal-hydraulic effects on ECCS and CSS considering the impacts of debris, focused on RELAP5-3D for small and medium hot-leg breaks. The licensee used alternative methods to evaluate cold-leg breaks, and large hot-leg breaks for the in-vessel thermal-hydraulic evaluations. A summary of these different methods is shown below in Table 3, "Accident Scenarios."

The NRC staff discussed the licensee's use of the RoverD methodology to evaluate the impacts of debris on cold-leg breaks in Enclosure 3, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Nos. 212 and 198 to Facility Operating License Nos. NPF-76 and NPF-80" (Main SE). Main SE Section 3.4.2.7, "Debris Transport," provides the NRC staff's evaluation of the licensee's analysis showing that, for all sizes of cold-leg breaks, 7 grams per fuel assembly (gm/FA) was the maximum that could reach the core. The NRC staff found this analysis acceptable because this debris amount met the WCAP-16793 criteria. The SE for WCAP-16793 provides the NRC staff's conclusion that debris quantities of less than 15 gm/FA will not result in core inlet blockage compromising core coolant flow.

For large hot-leg breaks, in order to simplify the thermal-hydraulic analysis, the licensee assumed that core damage would result for any hot-leg break greater than 16 inches, thus, a risk-informed assessment (probabilistic risk assessment) was used. Table 3 provides an overview of the different break scenarios and how they are treated, and shows that only small and medium hot-leg breaks are considered in this in-vessel thermal-hydraulic analysis.

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[4] A figure of merit is calculated during the analysis and is a primary variable for drawing conclusions about the analysis. An analysis may have several figures of merit.

Table 3: Accident Scenarios

	Hot-leg Break	Cold-leg Break
Small Break	LTCC EM Analysis (In-Vessel Thermal-Hydraulic Analysis)	RoverD Analysis (Main SE)
Medium Break	LTCC EM Analysis (In-Vessel Thermal-Hydraulic Analysis)	RoverD Analysis (Main SE)
Large Break (> 16")	Risk-informed Analysis (Main SE)	RoverD Analysis (Main SE)

A.3.3 System Description

The facility is a four-loop PWR with a Westinghouse-designed nuclear steam supply system. During a LOCA, regardless of break location, the ECCS pumps are aligned to inject borated water into three of the four RCS cold-legs. The source for this water is from stored locations, like the refueling water storage tank (RWST). Water that is pumped into the reactor vessel by the ECCS is subsequently discharged through the break into the containment where it collects in the containment building basement and the ECCS sumps. When the stored water supply is exhausted, the CSS and ECCS are realigned to draw coolant from the containment sump. The coolant discharged from the RCS and from the CSS is then circulated back into the RCS to provide for continued LTCC without the need for additional cooling water.

There are two separate categories of LOCAs depending on whether the break is upstream or downstream of the core (cold-leg side or hot-leg side, respectively). The quantity of debris carried into the core, the quantity of debris deposited on fuel cladding surfaces, and the head available to drive coolant into the core are greatly dependent upon the location of the pipe break. The effect of the different break locations is discussed in WCAP-16793.

In the event of a hot-leg break, the coolant pumped into the cold-leg is forced into the reactor pressure vessel, down the downcomer and up through the reactor core toward the break. During the LTCC period, core flow, plus a small amount of core bypass flow, is equal to the total ECCS flow delivered to the cold-leg. However, this ECCS flow, and thus the flow through the core, may vary depending on the number of operating ECCS pumps.

In the event of a cold-leg break, ECCS coolant injected into the failed loop will exit the RCS through the break while coolant injected into the intact loop will enter the downcomer annulus. This ensures that the downcomer is filled, at minimum, to the bottom of the cold-leg nozzle. During a cold-leg break, once the core has been recovered, the flow of coolant entering the core is that required to replenish boil-off (i.e., less than 1.5 gallons per minute per fuel assembly) (Reference 4). The excess coolant flows around the downcomer annulus and exits the reactor pressure vessel through the failed pipe. Therefore, the LTCC period following a cold-leg break represents a minimum core flow condition.

Debris build-up at the core inlet and in the fuel assemblies following either a hot-leg or cold-leg break could impact heat transfer from the fuel cladding and could add to the resistance in the core inlet that must be overcome to provide adequate cooling flow into the core.

A.4 TECHNICAL EVALUATION

A.4.1 Clad Oxide Calculation

Clad Oxidation

The thickness of the cladding oxide and the deposits of material on the fuel shall not exceed 0.050 inches in any fuel region.

Acceptance Basis from WCAP-16793-NP, Revision 2

The clad oxide thickness limit provided in WCAP-16793 prevents deposits from filling the space between adjacent fuel rods and blocking coolant flow.

The licensee provided justification that it met this criterion in response to SNPB-3-1 (Reference 16). In that response (Reference 27), the licensee referenced responses to previous questions 31 and 36 which referred to a LOCA Deposition Model (LOCADM) analysis demonstrating that the total thickness of deposition remained less than 50 mils. However, it was not clear to the NRC staff that this LOCADM analysis was applicable to STP, as it appeared to assume a much lower fiber loading than could be justified for STP.

Therefore, the licensee supplemented its response to SNPB-3-1 (Reference 19) and provided additional details on the STP-specific analysis performed. The analysis performed with LOCADM assumed that 91 grams of fibrous debris per fuel assembly (gm/FA) bypassed the sump strainers. This was much greater than the amount that could be reasonably expected to bypass the strainer since the maximum amount that could bypass the strainer and still be considered a success at the strainer was less than half this amount. Because the licensee performed a plant-specific analysis that conservatively assumed a larger quantity of fiber per fuel assembly than would actually be expected for the scenarios under consideration, the NRC staff determined that LOCADM is appropriate for STP.

The licensee used LOCADM with a conservative amount of debris bypassing the sump strainers and confirmed that the thickness of clad oxide and deposits of material was less than 0.050 inches. The NRC staff determined that there is reasonable assurance that the clad oxidation will not exceed 0.050 inches in any fuel region because the licensee used LOCADM with a conservative amount of debris in the analysis. Therefore, the NRC staff concludes that the clad oxidation criterion is satisfied.

A.4.2 Long-Term Core Cooling Evaluation Model

As stated previously, the NRC staff's review of the EM was performed following the guidance of SRP Section 15.0.2. Section 15.0.2 of the SRP directs the reviewer to examine the EM, which is defined as the calculational framework for evaluating the behavior of the RCS during a postulated accident or transient, and includes the computer programs, mathematical models, assumptions, and procedures on how to treat the input and the output, as well as many other factors. Section 15.0.2 of the SRP organizes the review into the six categories shown in Table 4.

Table 4: SRP Section 15.0.2 Review Categories

Section	
A.4.2.1	Accident Scenario Identification Process
A.4.2.2	Documentation
A.4.2.3	Evaluation Model Development
A.4.2.4	Code Assessment
A.4.2.5	Uncertainty Analysis
A.4.2.6	Quality Assurance Program

In order to demonstrate that the maximum clad temperature will not exceed 800 °F following the core quench and reflood stage, STPNOC performed multiple simulations using the LTCC EM. The LTCC EM uses the RELAP5-3D computer code. The focus of the NRC staff's review was to determine if the code options, inputs, and models were appropriate and if using appropriate input parameters resulted in an accurate and conservative RELAP5-3D simulation.

The NRC staff's review focused on ensuring the LTCC EM simulations performed by STPNOC were reasonable representations of the actual thermal-hydraulic phenomenon in the STP core following a LOCA. The NRC staff's review of the EM was restricted to those simulations already performed by STPNOC, and did not consider future simulations which could be performed except for very small modifications to the initial set of three simulations. The NRC staff therefore placed limitations on the use of the LTCC EM by STPNOC to perform the analysis using alternative inputs. These limitations are discussed in Section A.4.3, "Conclusions."

A.4.2.1 Accident Scenario Identification Process

The accident scenario identification process is a structured process used to identify the key figures of merit or acceptance criteria for the accident. It is also used to identify and rank the reactor component and physical phenomena modeling requirements based on their (a) importance to acceptable modeling of the scenario and (b) impact on the figures of merit for the calculation (e.g., PCT and maximum, average cladding oxidation thicknesses).

In general, STPNOC considered six similar accident scenarios. However, only two scenarios were assessed deterministically and used its LTCC EM. The other four scenarios were addressed by other means and were not considered in this review. A summary of the accident scenarios and the SE sections where the NRC staff review is documented is shown in Table 3.

For the LTCC EM, the key figure of merit is the PCT. However, other calculated quantities (e.g., mass flow rates, pressures, heat fluxes) are important for ensuring that the resulting simulation behaves in a reasonable manner and that the resulting post-reflood PCT has been adequately simulated. Therefore, the focus of the accident scenario identification process is to describe the important phenomena in each scenario, so the EM can be evaluated in terms of its ability to model those phenomena. Table 5 provides the SRP review criteria topics and the sections providing the NRC staff's review.

Table 5: Accident Scenario Identification Process Review Categories

Accident Scenario Identification Process	
A.4.2.1.1	Structured Process
A.4.2.1.2	Accident Progression
A.4.2.1.3	Phenomena Identification and Ranking
A.4.2.1.4	Initial and Boundary Conditions

A.4.2.1.1 Structured Process

Structured Process

The process used for accident scenario identification should be a structured process.

SRP Section 15.0.2, Subsection III.3c

The licensee provided a description of the structured process used to identify and define the accident scenario in response to SNPB-3-2 (References 18 and 19) and SNPB-3-4 (Reference 16). The NRC staff's review of the STPNOC process for accident scenario identification determined that the process addressed three areas:

1. The description of the accident scenarios; evaluated by the NRC staff in Section A.4.2.1.2, "Accident Progression," of this In-Vessel Thermal-Hydraulic Analysis.

2. Identification of the important phenomena from these scenarios; evaluated by the NRC staff in Section A.4.2.1.3, "Phenomena Identification and Ranking," of this In-Vessel Thermal-Hydraulic Analysis.

3. Identification of the important aspects of the boundary conditions; evaluated by the NRC staff in Section A.4.2.1.4, "Initial and Boundary Conditions," of this In-Vessel Thermal-Hydraulic Analysis.

Because the licensee considered the accident scenario, identified the important phenomena, and identified the important boundary conditions, the NRC staff determined that the process used for the accident scenario identification is a structured process; therefore, this criterion is satisfied.

A.4.2.1.2 Accident Progression

Accident Progression
The description of each accident scenario should provide a complete and accurate description of the accident progression.
SRP Section 15.0.2, Subsection III.3c

In its response to SNPB-3-2 (References 18 and 19), the licensee provided a description of the accident progression for the largest break considered deterministically as well as justifications for certain initial and boundary conditions. The justifications related to the input and boundary conditions are evaluated in Section A.4.2.1.4, "Initial and Boundary Conditions," of this In-Vessel Thermal-Hydraulic Analysis.

The main phases of the 16-inch hot-leg break: break and blowdown, refill and reflood, pre-blockage and LTCC, and core blockage and post-blockage LTCC, roughly follow those of the standard cold-leg break LOCA accident progression, with some exceptions. The phases of the 16-inch hot-leg break are described below. The following STP information should be noted for this break progression:

- Both units at STP are 4-loop Westinghouse plants, each with three independent trains of safety injection (SI).
- Safety injection flow is injected into the RCS in loops A, B, and C between the steam generator and the core in the cold-leg.
- The break is located in loop B between the core and the steam generator in the hot-leg.

<u>Phase 1 - Break and Blowdown</u>

The licensee stated that the first phase is defined to start when the hot-leg breaks and the reactor depressurizes as water is expelled from the RCS through the 16-inch break. Since the water's temperature is above the saturation temperature as the RCS blows down, depressurization causes voids to form in the RCS. However, there is substantial coolant flow through the core, since all SI flow must pass through the core to exit out of the break in the hot-leg. Because of this, the core collapsed liquid level never drops below the bottom of the core.

<u>Phase 2 - Refill and Reflood</u>

The licensee continued with the second phase stating that it starts when the core collapsed liquid level starts increasing and ends when the core is completely flooded (i.e., the collapsed liquid level reaches the top of the core). Because this is a hot-leg break (where SI must flow through the core to reach the break) rather than a cold-leg break (where the SI can bypass the core), the licensee stated that the beginning and end of this phase can be difficult to identify. During this phase, the accumulators inject and eventually the low head SI pumps are able to complete the flooding of the core.

Phase 3 - Pre-Blockage and Long-term Core Cooling

The licensee stated that the third phase is defined to start when the core is completely flooded (i.e., the collapsed liquid level has reached the top of the core) and ends when the ECCS suction switches over from the RWST to the sump. During this time, the SI flow completes filling up the reactor pressure vessel and RCS loops and starts filling up the steam generators. The steam generators, which were unable to release their stored energy to the RCS during the first three phases because of voiding on the primary side of the steam generator u-tubes, act as heat sources that must be cooled by the SI flow. In its supplemental response submitted in Reference 19, STPNOC asserted that the large quantity of secondary-side mass in the steam generators, which consists of both water and metal mass, did not interfere with the refilling of the steam generator u-tubes. In the sensitivity study STPNOC provided to support this assertion, conservatively large masses were added to the secondary side of the steam generator which more than accounted for the secondary metal mass in the steam generators. Even with these additional masses and the increased energy transferred to the SI flow, there was no impact on the PCT for the LTCC EM.

Phase 4 - Core Blockage and Post-Blockage Long-Term Core Cooling

The licensee described phase 4 as when the SI flow switches over from the RWST to the sump, thus causing any debris which accumulated in the sump and bypassed the sump strainer to be pumped into the core. The licensee assumed the time between sump switchover and core blockage is 360 seconds for this analysis. The justification for this time delay is evaluated by the NRC staff in Section A.4.2.1.4, "Initial and Boundary Conditions," of this In-Vessel Thermal-Hydraulic Analysis.

After 360 seconds, the debris generated by the LOCA and transported to the RCS is conservatively assumed to completely block the bottom of the core, including the barrel-baffle bypass region. In reality, as the core begins to block, a portion of the SI flow would be diverted around the core and into the steam generator u-tubes. The fraction of flow diverted around the core increases as a function of core blockage until the bottom of the core is completely blocked, at which point all of the SI flow is forced through the steam generator u-tubes. However, for the simulation, this gradual blockage is not assumed and instead the licensee assumes the core is instantaneously blocked at 360 seconds following sump switchover.

Following sump switchover, STPNOC specified all SI must flow through one of the steam generators. Any SI flowing to steam generator 3 (i.e., the loop with the break) must flow in through the cold-leg of the steam generator, through the u-tubes, down the hot-leg and exit the RCS on the steam generator side of the break. Any SI flowing to steam generators 1, 2, or 4 must flow in through the cold-leg of the steam generator, through the u-tubes, down the hot-leg, and into the core to the upper plenum before it is able to flow into the hot-leg of loop 3 and out the vessel side of the break.

Following blockage, the temperature of the fluid in the core increases and boiling may occur. Fluid will then flow from the core into the upper plenum due to the change in density, caused either by a phase change to steam or by the decreasing density of liquid water as the temperature increases. This fluid, likely a two-phase mixture, will combine with the SI flowing into the upper plenum from the steam generators and some of the mixture will flow back into the core through the upper core plate. Eventually, because of the coolant added to the system by the ECCS, the mixture must flow into the hot-leg of the broken loop and exit the RCS on the vessel side of the break.

The NRC staff reviewed the licensee's description of each phase of a hot-leg break LOCA that included the key physical phenomena, the key figures of merit, and the progression of each during the phase. This description was logical and consistent with the NRC staff's experience with these phases of the event. The NRC staff concludes that the licensee provided a complete and accurate description of the accident progression; therefore, this criterion is satisfied.

A.4.2.1.3 Phenomena Identification and Ranking

Phenomena Identification and Ranking

The dominant physical phenomena influencing the outcome of the accident should be correctly identified and ranked.

SRP Section 15.0.2, Subsection III.3c

The licensee provided justification for the above criterion in response to SNPB-3-4 (Reference 16). It should be noted that at the time of this response, the licensee intended to use the LTCC EM for small, medium, and large (i.e., larger than 16-inch breaks) hot-leg breaks as well as small cold-leg breaks. Therefore, the licensee provided a description of the important phenomena for those set of breaks. Only the phenomena associated with hot-leg breaks 16 inches and smaller were considered for this review since STPNOC changed its request to make all breaks larger than 16 inches to be part of the risk-informed review (the NRC staff's review of this is provided in the Main SE).

The licensee's listing of phenomena was based on work performed under the code scaling, applicability, and uncertainty (i.e., CSAU) methodology to identify the important phenomena occurring during a large break LOCA (LBLOCA) (Reference 29). The listing included a description of the important phenomena and a discussion of the phase of the accident during which each phenomenon occurred. The NRC staff found that the phenomena identified encompassed all phenomena the staff would expect to be important during a hot-leg break.

The licensee also identified those phenomena expected to be most important during the long-term cooling phase (phase 4). The NRC staff determined that the licensee's identification of both heat transfer of natural convection and the counter current flow limitation (CCFL) would be the most important phenomena during the long-term phase. However, the NRC staff also identified other phenomena which could be important during phase 4, including:

- Blockage on grid spacers in the fueled regions of the core
- Core uncovery
- Heat stored in the steam generators

<u>Blockage on grid spacers in the fueled region of the core</u>

The licensee's analysis assumed some amount of debris makes it past the strainer and blocks the bottom of the core, so it is reasonable to assume that the SI flow has the potential to carry some amount of debris into the core. This debris could cause additional blockages in the core, resulting in local heat ups. The licensee provided a justification that debris in the fueled region of the core was not an important consideration in response to SNPB-3-5 (Reference 16).

In its response, the licensee referenced a previous submittal (Reference 30). In the referenced analysis, the licensee demonstrated that the amount of crud which could be expected to be released would be less than 2 pounds. The NRC staff reviewed the licensee submittal and determined that this small amount of fine particles would have minimal impact on the flow through the grid spacers. However, though crud deposition is an important consideration, the analysis discussed by the licensee did not address the potential for fibrous debris deposition in the core.

In its response (Reference 16), the licensee also referenced WCAP-16793 to address the ability of fibrous debris to collect at grid spacers. In Section 3.4.4 of the SE for WCAP-16793, the NRC staff concluded that when the quantity of debris is within the acceptance limits specified in WCAP-16793 (e.g., 15 gm/FA), there is reasonable assurance that flow of coolant will not be impeded by debris collecting at grid spacers. This conclusion was based on the observation during testing that while fiber was deposited on grid spacers in the core, it did not impede cooling. It was not clear to the NRC staff that the same conclusion made in WCAP-16793 would be applicable to the licensee, since the expected quantity of fiber at STP is higher than what was tested in WCAP-16793.

In its supplemental response to SNPB-3-5 submitted in Reference 19, the licensee discussed the potential for debris beyond 15 gm/FA to enter the core. The licensee described the following conservatisms in its modeling approach:

- The maximum amount of fiber that is available to reach the core is estimated as a conservatively high amount of 50 gm/FA. This amount is expected to be conservative as debris greater than 50 gm/FA could only occur if the sump strainer fails.

- The core is assumed to be 100 percent blocked as soon as the in-vessel fiber amount reaches 15 gm/FA. As demonstrated by testing, a much larger fiber loading would be needed to fully block the bottom of the core provided chemicals have not arrived. Additionally, once fiber begins to block the bottom of the core, it forms a bed which acts as a filter and easily traps additional fiber. This leads to the conclusion that the amount of fiber which would be trapped at the bottom of the core would likely be well in excess of 15 gm/FA, reducing the fiber amount available for deposition in the fueled region of the core.

- Any SI flow which is discharged from the core following sump switchover would be re-filtered through the ECCS sump screens, which would reduce the amount of fiber injected into the vessel. This re-filtering is ignored in the current analysis.

- The testing which supports the 15 gm/FA was conducted in such a way that any re-filtering of the coolant through the ECCS suction strainers was also ignored. If this was accounted for, the licensee stated the actual amount of fiber would have been greatly reduced due to the SI being re-filtered through the ECCS sump strainers.

Assuming that the maximum amount of fiber entering the core would be 50 gm/FA, and 15 gm/FA would be deposited on the core support plate, 35 gm/FA would remain to be deposited in the core above the bottom grid, deposited elsewhere in the RCS, or flow out of the break. The licensee noted that only a small portion of this flow would actually enter the core, since the only SI flow to enter the core would be the flow to overcome decay heat. Much of the SI flow would exit the break, thus, the licensee estimates that only a small amount of debris, about 5 gm/FA, would flow from the upper plenum, through the upper core plate, and into the core. However, the NRC staff considers the flow dynamics to be complicated by the conservative core blockage assumptions in the analysis as discussed below.

While it is conservative to assume that the core is fully blocked once the debris entering the vessel reaches 15 gm/FA, this means that, for a hot-leg break, the only other flow path is around the steam generators, through the hot-legs, and into the upper plenum. As only a small fraction of the SI flow will be needed to replace that which is pushed out of the core due to natural convection, only a small amount of debris would actually enter the core. While that scenario is conservative in that it reduces the amount of SI flow into the core, this reduction in SI flow also reduces the amount of debris capable of entering the core, therefore, the NRC staff does not consider this scenario to be likely.

The NRC staff considers it more likely that the core will not block with 15 gm/FA deposited on the core support plate; and further, that there will be substantial flow through the barrel-baffle region, which also will not block. While the core is generally assumed to block below the core support plate, the blockage would actually occur below the first spacer. It is reasonable to assume that some amount of fiber would be deposited on these grid spacers, but it is likely that the distribution would not be uniform on all grid spacers and all grid spacers may not block. If blockage occurred uniformly on all grid spacers, it would likely not happen simultaneously across the core. Further, there is no known blockage mechanism in the barrel-baffle region as the holes are too large for the fiber to bridge across. The barrel-baffle region not only provides a flow path directly to the top of the core, it also provides the capability for coolant to be delivered into the core at various axial elevations because of the horizontal flow holes in the core baffle, which are also likely to remain unblocked by fibrous debris. Thus, there is a reasonable probability that much more of the SI flow (and hence, the associated debris) would actually enter the vessel than assumed in the "conservative" case.

The NRC staff considered both extremes wherein either (a) many of the flow channels are blocked and only a very small amount of SI flow would be expected to enter the fueled region of the core, or (b) few of the flow channels are blocked and a large amount of SI would be expected to enter the fueled region of the core. The staff finds that, given the open-lattice configuration of a PWR core which allows for multiple paths of cross flow and the analysis suggesting that the core region would be very well mixed, there is reasonable assurance that any fiber deposited in the fueled region of the core will have a minimal impact on core heat transfer. In situation (a), a minimal amount of fiber is able to flow into the core. In situation (b), a large amount of fiber will flow into the core, but this fiber will be carried by a large amount of SI flow which would aid in mixing the fluid in the core and prevent localized heating.

<u>Core Uncovery</u>

As long as the core remained covered with a two-phase mixture, then the core temperature would be kept near saturation and the heat transfer would be limited to two-phase convection. However, following LTCC, should the two-phase level drop below the top of the core, the fuel would experience a heat-up to temperatures greater than saturation, and other heat transfer regimes should be considered as highly-ranked phenomena. In the license's base case, the two-phase level never dropped below the top of the core. However, the base case assumed a cosine axial power shape. In response to SNPB 3-20 (Reference 18), the licensee demonstrated that, following sump switchover and core blockage, both the top-skewed and bottom-skewed axial power shapes would result in higher PCTs than the cosine axial power shape. Further, in both instances, the PCT rose above the saturation temperature in the core. This suggested that the limiting case should assume a top-skewed axial power shape, as the sensitivity study showed that it reached the highest post-reflood PCT. Additionally, in that study, the two-phase level dropped below the top of the core, making heat transfer regimes other than the ones initially considered by the licensee highly ranked phenomena. It should be noted that in all cases, the PCT did not remain above saturation temperature for an extended period (i.e., it was a transient effect which did not last long enough to significantly impact temperature) of time, and the cladding temperature remained well below the limit of 800 °F.

In a supplemental response to SNPB 3-20 (References 19 and 20), the licensee provided additional details to support the assumption that the core would not experience temperatures above saturation, and, therefore, consideration of other heat transfer phenomena was not necessary. The main argument provided by the licensee was that the assumption of complete core blockage was conservative. Testing conducted by Westinghouse[5] demonstrated that the holes of the barrel-baffle region would not block due to the larger size of the holes and the velocity of the SI flow through them. The licensee performed a sensitivity study assuming a top-skewed axial power shape with the barrel-baffle flow path opened. This sensitivity study demonstrated that the core remained at saturated conditions following sump switchover and core blockage. The NRC staff found that, given the scenario with the barrel-baffle region open is the most realistic and supported by test data, it would be very unlikely that the core would experience temperatures above saturation. Additionally, this study did not model the horizontal flow holes connecting the barrel-baffle region and the core, which is an additional conservatism. Modeling these holes would further increase the SI flowing into the core and improve heat transfer, further reducing the likelihood of observing temperatures above saturation in the core.

Heat stored in steam generators

The licensee stated that the energy stored in the steam generators could impact the PCT in two main ways. First, the energy transferred to the primary side could cause boiling, which would increase the pressure drop and consequently increase the time it takes to fill a steam generator. Second, the energy transferred to the primary side could cause a reduction in the subcooling of the SI flow before it has the opportunity to cool the core. The licensee provided justification that the heat stored in the steam generators was not an important consideration in response to SNPB-3-2 (Reference 19). In its response, the licensee performed a sensitivity study and increased the secondary side metal mass by accounting for the mass of the u-tubes twice. This resulted in increasing in the total heat transferred from the steam generators to the primary side by a factor of 6. Even with this large increase in heat transferred, the resulting LTCC PCT remained within a small range of the saturation temperature. Given this sensitivity study, the NRC staff concludes that the heat stored in the steam generators has minimal impact on the PCT.

In summary, the licensee provided a list of the phenomena, a description of those phenomena, and the justifications of ranking those phenomena. The NRC staff reviewed this information and, as discussed above, determined that the dominant physical phenomena of "core uncovery" and "blockage on grid spacers in the fueled regions of the core" were ranked above "heat stored

[5] See WCAP-17788, Volume 1, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090)," Section 6.4.1; and Volume 5, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090) - Autoclave Chemical Effects Testing for GSI-191 Long-Term Cooling," Section 5.6.

in steam generators." Thus, the phenomena have been identified, described, and ranked; therefore, this criterion is satisfied.

A.4.2.1.4 Initial and Boundary Conditions

Initial and Boundary Conditions
The description of each accident scenario should provide complete and accurate description of the plant initial and boundary conditions.
SRP Section 15.0.2, Subsection III.3c

The licensee provided information on this criterion in response to SNPB-3-6 (Reference 18). Specifically, the licensee provided the key parameters assumed in its simulations and confirmed these parameters with a direct comparison to plant data, or, if plant data were unavailable, with the results from a steady state run of the approved plant transient analysis code RETRAN. Though the licensee's verification assured the accuracy of most of the key initial and boundary conditions, the NRC staff asked questions about the following parameters:

- Treatment of core bypass flow
- Time from sump switchover to full-core blockage (360 seconds)
- Bounding break size (16-inch)
- Initial and boundary conditions for phase 4 - post-blockage LTCC

Core bypass flow

During the review, the licensee did not clearly indicate whether it credited the bypass flow in the barrel-baffle region or other core bypass flow paths. The licensee provided a clarification in response to SNPB-3-3 (Reference 16) by discussing the six flow paths which constitute the core bypass flow, but did not specify how each would be treated. In its supplemental response to SNPB-3-22 (Reference 18), the licensee addressed how each of the core bypass flow paths were modeled, and clarified that the flow in the barrel-baffle region is conservatively assumed to be blocked once the core is blocked at the time of sump switchover. While the barrel-baffle bypass flow is not credited in the official base-case, a sensitivity study was performed in response to SNPB-3-20 (Reference 20), which showed the impact of allowing flow in the barrel-baffle region. Based on the foregoing information, the NRC staff finds that the licensee's description of the core bypass flow paths is complete and accurate.

Time from sump switchover to full-core blockage (360 seconds)

The licensee provided justification that the time from sump switchover to full-core blockage was conservative in its response to SNPB-3-2 (Reference 19). In that response, the licensee clarified that the time between sump switchover and core blockage of 360 seconds corresponds to the time needed for 15 gm/FA to enter the RCS following sump switchover. In the analysis, this blockage is applied instantaneously at 360 seconds; it does not build up gradually, as would be expected in reality. The NRC staff finds this treatment of blockage timing conservative for two reasons.

First, the assumption that the core fully blocks once 15 gm/FA collects at the core inlet is a conservative assumption. As previously discussed, testing was used to demonstrate that

15 gm/FA applied at the core inlet would not significantly impede core cooling. The testing did not quantify how much fiber would be necessary to fully block the core, though testing demonstrated that, without chemical effects, a much higher value than 15 gm/FA could be supported.

Second, in reality, as the fiber layer started to build up on the lower spacers in the core, the pressure drop through the core would gradually increase before the core inlet became fully blocked. This would divert flow through other paths, including the steam generators. Thus, not all of the SI flow entering the RCS would actually make it to the core inlet, and once the RCS reached 15 gm/FA the amount of fiber at the core inlet would likely be less than that amount.

For the two reasons discussed above, the NRC staff finds the licensee's assumption of 360 seconds from sump switchover to full-core blockage as reasonable. Thus, the NRC staff finds the licensee's description of the sump switchover time is complete and accurate.

Break size (16-inch)

The licensee provided a justification that the 16-inch hot-leg break bounds all smaller breaks in response to SNPB-3-2 (Reference 19). In its response, the licensee noted that unlike medium and small cold-leg breaks where PCT is not correlated to break size, in medium and small hot-leg breaks, PCT is correlated to break size, with higher PCTs occurring at larger break sizes. This is because in smaller breaks, the pressure remains high for longer and ECCS flow is injected at a slower rate, allowing the RWST to drain slower and delaying the time before sump switchover. Therefore, smaller breaks will result in a lower decay power when sump switchover occurs and will also result in a slower buildup of debris in the RCS, since the SI injection would be ultimately limited to the break flow.

The licensee also performed a sensitivity study for three break sizes (16 inches, 6 inches, and 2 inches). This sensitivity demonstrated that the largest break size had the earliest sump switchover and therefore the highest decay power at core blockage. The licensee noted that the same four phases (break and blowdown, refill and reflood, pre-blockage and LTCC, and core blockage and post-blockage LTCC) are experienced in each case, with the break size mostly impacting the length of each phase. However, the licensee recognized that the figure of merit for determining the limiting break should not be PCT, but the core collapsed liquid level. This is because, in the analysis, the fuel remains covered and the PCT is therefore close to the saturation temperature. Thus, for an event such as the 2-inch break where the core never completely depressurizes, the saturation temperature remains high. The NRC staff reviewed the scenario and determined that the most limiting break would be the one resulting in core uncovery in the long-term phase with the highest PCT (or the break which brings the system closest to core uncovery during the long-term phase). Therefore, the limiting hot-leg break is the 16-inch break since it bounds all smaller breaks in the hot-leg piping. The NRC staff finds that the licensee's description of the bounding break size to be complete and accurate.

Initial and boundary conditions for phase 4 - post-blockage long-term core cooling

The licensee used the LTCC EM to simulate the entire event, however, the NRC staff considered the first three phases of the accident – break and blowdown, refill and reflood, pre-blockage and LTCC – only inasmuch as they provided the initial and boundary conditions for the fourth phase, post-blockage long-term core cooling. In other words, the NRC staff did not review the ability of the EM to adequately capture the PCT during blowdown or reflood, since the core is completely quenched at the start of the phase and the PCT would have no impact on the fourth phase. Instead, the NRC staff considered which aspects of the modeling of the first three phases would have the largest impact on the fourth phase.

The licensee stated that because the sump switchover occurred after reflood, the conditions at the beginning of phase 4 were relatively constant. In general, the core would be between the ECCS injection temperature and saturation temperature. The NRC staff found three areas in which the previous phases could have an impact on the fourth phase: the delay after full-core blockage when SI flow reaches the core, the decay power in the core at the time of blockage, and the amount of subcooling in the core.

- Concerning the delay after full-core blockage: The NRC staff noted that the licensee conservatively increased the delay after full-core blockage when SI flow reached the core by ignoring the gradual build-up of debris and ignoring most alternate flow paths. In reality, the debris would gradually build up on the bottom-most assembly spacer grid, which would result in a higher pressure drop through the core. This higher pressure drop would result in more flow being diverted to the steam generators. Because the steam generators become effectively the only path for SI flow to enter the core once the core inlet is blocked, the primary side of the steam generator u-tubes must be completely full to establish core cooling. In the current analysis, the steam generator u-tubes do not completely fill with water until blockage occurs, because the major flow path is through the core and out the break. This increases the delay between core blockage and when SI flow can reach the core. If the debris build-up were modeled as a gradual increase instead of a step change, additional flow would be diverted to the steam generators, causing them to fill faster and reducing the delay between core blockage and when SI flow reaches the core. Further, as detailed in response to SNPB-3-3 (Reference 16), the licensee ignored alternate flow paths which would also decrease this time delay and could prevent a time delay altogether. Therefore, the licensee made conservative assumptions concerning the delay after full-core blockage.

- Concerning the decay power in the core at the time of blockage: The NRC staff noted that the licensee ensured a conservatively early time for full-core blockage by reducing the time to sump switchover. This was accomplished by minimizing volume in the RWST and assuming all SI trains were fully operational and not operating at minimal efficiency. Since the licensee reduced the time to sump switchover and blockage, decay heat in the core would be higher requiring a higher cooling capacity. Therefore, the licensee made conservative assumptions concerning the decay power in the core at the time of blockage.

- Concerning the amount of subcooling in the core: The NRC staff noted that the licensee conservatively reduced the subcooling in the core at the start of the fourth phase. This was accomplished by maximizing the RWST temperature and ignoring any ECCS cooling. Since the licensee assumed lower cooling capacity of the RWST coolant and no ECCS cooling, the licensee conservative treated the amount of subcooling in the core.

The NRC staff concludes that the licensee (1) applied initial and boundary conditions that reflect plant operating conditions, (2) used justifiable initial and boundary conditions for key inputs such as break size and time to sump switchover, and (3) modeled the first three phases of the event in such a way as to result in conservative initial conditions for the fourth phase. The fourth phase conservative assumptions included increasing the time delay between full-core blockage and when SI flow reaches the core, ensuring an early blockage time with a resulting higher heat load in the core, and reducing the subcooling in the core. The NRC staff therefore determined that the licensee provided complete and accurate descriptions of the plant initial and boundary conditions, and that the initial and boundary conditions for the accident scenario reflect real or conservative plant conditions, thus, the NRC staff concludes that this criterion is satisfied.

A.4.2.2 Documentation

The development of an EM for use in reactor safety licensing calculations requires a substantial amount of documentation including (a) the EM, (b) the accident scenario identification process, (c) the code assessment, (d) the uncertainty analysis, (e) a theory manual, (f) a user manual, and (g) the Quality Assurance Program (QAP).

Section 15.0.2, Subsection III.3.a of the SRP contains seven review criteria for the NRC staff's documentation assessment. The review criteria topics and subsections with the NRC staff's review criteria assessments are listed in Table 6.

Table 6: Documentation Review Categories

Documentation	
A.4.2.2.1	Necessary Documentation
A.4.2.2.2	Theory Manual
A.4.2.2.3	Closure Relationships
A.4.2.2.4	User Manual
A.4.2.2.5	Options for Licensing Calculations
A.4.2.2.6	Required Input
A.4.2.2.7	Accident-Specific Guidelines

A.4.2.2.1 Necessary Documentation

Necessary Documentation

The documentation should be reviewed to determine if (i) all documentation listed in Section II.1 above has been provided [the evaluation model, the accident scenario identification process, the code assessment, the uncertainty analysis, a theory manual, a user manual, and the quality assurance program], (ii) the evaluation model overview provides an accurate roadmap of the evaluation model documentation, (iii) all documentation is accurate, complete, and consistent and, (iv) all symbols and nomenclature have been defined and consistently used.

SRP Section 15.0.2, Subsection III.3a

In Chapter 5 of Attachment 1-3 to Supplement 2 (Reference 10), STPNOC provided an initial overview of the LTCC EM. Additional documentation, including the licensee's changes to the LTCC EM, is captured in the following list:

- RELAP5-3D code manuals (Reference 9)

- Relevant STPNOC RAI responses pertaining to the LTCC EM:

    - RAI Response – Part 1 (Reference 16)
    - RAI Response – Part 2 (Reference 17)
    - RAI Response – Part 3 (Reference 18)
    - RAI Response – Supplement (Reference 19)
    - RAI Response – Supplement (Reference 20)

Based on the NRC staff's review of the above references, the licensee provided all documentation necessary for the NRC staff to complete its documentation review, the licensee's EM overview provides an accurate roadmap of the EM documentation, the licensee's EM is adequately described for the simulations performed and the documents were accurate, complete, and consistent, with all symbols and nomenclature defined and used consistently. Therefore, the NRC staff concludes that this criterion is satisfied.

A.4.2.2.2 Theory Manual

Theory Manual

The theory manual should be a self-contained document that describes the field equations, closure relationships, numerical solution techniques, and simplifications and approximations (including limitations) inherent in the chosen field equations and numerical methods.

SRP Section 15.0.2, Subsection III.3a

The licensee provided information on the theory manual in response to SNPB-3-8 (Reference 16). In the licensee's response, it provided a brief description of each key phenomena, how each phenomenon was modeled, and where more information on each phenomenon can be found in the RELAP5-3D theory manual (Reference 9).

The licensee provided the theory manual for RELAP5-3D, a listing of the important phenomena and the link between those phenomena and how they are modeled in the LTCC EM, and other docketed correspondence describing the inputs and other assumptions (References 16, 17, 18, 19, and 20).

The NRC staff determined that the theory manual for RELAP5-3D, and any particular input, model, and parameter selections resulting from its use in the LTCC EM as specified in the RAI responses, is a self-contained document containing the field equations, closure relationships, numerical solution techniques, and associated simplifications and approximations for these equations and techniques. The NRC staff concludes the licensee provided an appropriate theory manual, and, therefore, this criterion is satisfied.

A.4.2.2.3 Closure Relationships

> **Closure Relationships**
>
> *The theory manual should identify the pedigree or origin of closure relationships used in the code and the limits of applicability for all models in the code.*
>
> SRP Section 15.0.2, Subsection III.3a

The licensee provided information on the closure relationships in response to SNPB-3-9 (Reference 17). In the licensee's response, it provided pointers to the relevant volume of the RELAP5-3D manual, which describes the various closure models for use in the code. This information, along with the phenomena mapping given in response to SNPB-3-8 and the detailed information provided in the RELAP5-3D manuals, allows the pedigree (i.e., history and origin) and limits (i.e., range of applicability) of each closure relationship to be determined.

The licensee provided the link between the important phenomena and their associated technical references in the RELAP5-3D manual, thus, the NRC staff finds that the closure relationships have been appropriately documented. Therefore, the NRC staff concludes that this criterion is satisfied.

A.4.2.2.4 User Manual

> **User Manual**
>
> *The user manual should provide guidance for selecting or calculating all input parameters and code options.*
>
> SRP Section 15.0.2, Subsection III.3a

The licensee provided information on the user manual in response to SNPB-3-10 (Reference 17). It should be noted that the user manual was provided early in the review process and describes four scenarios for its use. It was only after the submittal of the RAI response that the licensee decided to address the small cold-leg break using a different approach. The user manual given to the NRC staff by the licensee provides information on how to execute the input decks, rather than providing instructions for building an input deck as would normally be expected in a user manual for an EM. Thus, in and of itself, the user manual is of limited use for performing new or additional simulations using the same LTCC EM. However, since the scope of review was restricted to consider the LTCC EM and the simulations which were generated by STPNOC for use only by STPNOC, future use of LTCC EM under different conditions were not considered in this review.

Since the user manual is mainly important for future use of the LTCC EM, the NRC staff determined the user manual to be irrelevant for this review. By limiting the scope of the review to only those simulations described in the licensee's RAI responses, and by restricting the LTCC EM use to STP only, the NRC staff concludes that this criterion does not apply.

A.4.2.2.5 Options for Licensing Calculations

Options for Licensing Calculations

The guidance in the [user] manual should specify the required and acceptable code options for the specific licensing calculations.

SRP Section 15.0.2, Subsection III.3a

Because the LTCC EM is only used to perform the simulations described in the RAI responses provided by the licensee, the NRC staff determined that consideration of future licensing calculations was not needed. The NRC staff concludes that this criterion does not apply.

A.4.2.2.6 Required Input

Required Input

The required input settings are hardwired into the input processor so that the code stops with an error message if the required input is not provided or if the input is not within an acceptable range of values or that administrative controls (an independent reviewer QA check) are in place that accomplish the same purpose.

SRP Section 15.0.2, Subsection III.3a

The LTCC EM makes use of the RELAP5-3D computer code. The NRC staff obtained a copy of RELAP5-3D code and confirmed that the code would stop with error messages and warnings if certain input was not provided or if the simulation provided erroneous results. While it is not feasible to confirm that every error in input will result in an error message, the wide use of RELAP5-3D, the wide use of the computer code which is it based on (RELAP5/MOD3), and its further development provides confidence that many such potential input errors have been discovered and corrected. Additionally, the limited use of the LTCC EM, restricted to the simulations already performed and slight variations from these simulations, would likely not result in any new errors. Therefore, the NRC staff concludes the computer code, RELAP5-3D, input setting are hardwired into the input processor and will provide the appropriate error messages when the required input is not provided; therefore, this criterion is satisfied.

A.4.2.2.7 Accident-Specific Guidelines

Accident-Specific Guidelines

Computer codes that are used for multiple accidents and transients should include guidelines that are specific to each transient or accident.

SRP Section 15.0.2, Subsection III.3a

The licensee provided justification for the use of the LTCC EM for the simulations of the 16-inch hot-leg breaks (and various sensitivity studies), however, complete accident-specific guidelines were not provided. This is because the NRC staff approval is limited to only those simulations already submitted to the NRC. Thus, future use of the LTCC EM by this licensee or others requires prior review and approval by the NRC staff.

The NRC staff concludes that this criterion is met since accident specific guidelines were provided for the STP simulations performed and the licensee is not authorized to use the computer codes for accidents and transients outside the simulations reviewed by the NRC staff.

A.4.2.3 Evaluation Model Development

As discussed in Section A.2, "Regulatory Criteria," of this In-Vessel Thermal-Hydraulic Analysis, 10 CFR 50.46 defines an EM. Section 15.0.2, Subsection III.3.b of the SRP contains eight review criteria for EMs. The review criteria topics and the subsections that provide the NRC staff's assessments are listed in Table 7.

Table 7: Evaluation Model Review Categories

Subsection	
A.4.2.3.1	Previously Reviewed and Accepted Codes and Models
A.4.2.3.2	Physical Modeling
A.4.2.3.3	Field Equations
A.4.2.3.4	Validation of the Closure Relationships
A.4.2.3.5	Simplifying and Averaging Assumptions
A.4.2.3.6	Level of Detail in the Model
A.4.2.3.7	Equations and Derivations
A.4.2.3.8	Similarity and Scaling

A.4.2.3.1 Previously Reviewed and Accepted Codes and Models

Previously Reviewed and Accepted Codes and Models

It should be determined if the mathematical modeling and computer codes used to analyze the transient or accident should have been previously reviewed and accepted.

SRP Section 15.0.2, Subsection III.3b

The LTCC EM makes use of the RELAP5-3D computer code, which is the latest in the RELAP5 series of computer codes created by Idaho National Laboratory. While other computer codes based on RELAP5 have been submitted, reviewed, and accepted by the NRC staff (Reference 26), RELAP5-3D has not. Additionally, previous approvals of LOCA codes in EMs focused on the blowdown, refill, and reflood phases of a LOCA, rather than the LTCC phase.

The LTCC EM submitted by the licensee focuses on LTCC with debris. The NRC staff is not aware of any computer codes which have been specifically reviewed and accepted for LTCC analysis. Therefore, the review of the LTCC EM using RELAP5-3D is considered a new review and will not directly rely on the NRC's acceptance of previously reviewed computer codes. However, in certain instances, the licensee compared RELAP5-3D predictions to those of previously reviewed and accepted methods; each of these instances is addressed individually.

The NRC staff determined that the mathematical modeling and computer codes used to analyze the accident have not been previously reviewed and accepted. Therefore, the NRC staff reviewed those mathematical models and computers codes in this In-Vessel Thermal-Hydraulic Analysis. The NRC staff concludes that this criterion is satisfied.

A.4.2.3.2 Physical Modeling

Physical Modeling

The physical modeling described in the theory manual and contained in the mathematical models should be adequate to calculate the physical phenomena influencing the accident scenario for which the code is used.

SRP Section 15.0.2, Subsection III.3b

The licensee provided justification that the initial and boundary conditions were accurate in response to SNPB 3-8 and SNPB-3-11 (Reference 16). In those responses, the licensee provided a brief description of RELAP5-3D's capability to model certain physical phenomena. Additionally, the licensee provided a description of the accident progression (discussed in Section A.4.2.1.2, "Accident Progression") as well as an identification of the highly ranked phenomena (discussed in Section A.4.2.1.3, "Phenomena Identification and Ranking"). Because the focus of the LTCC EM was on the behavior of the core following sump switchover and full-core blockage, the NRC considered the physical modeling of the first three phases inasmuch as they provide reasonable initial and boundary conditions for the fourth phase. As detailed in the section on accident progression, the phenomena associated with the first three phases of the event are within the general scope of the standard LOCA analysis for which RELAP5-3D was developed. Additionally, the phenomena associated with phase 4 of the event are a subset of those phenomena which are considered important in the first three phases.

Because the scenarios under consideration are expected to experience the same phenomena as those which occur during a LOCA, and because RELAP5-3D was developed with models for all key LOCA phenomena (including those identified by the licensee and reviewed by the NRC staff as important in phase 4), the NRC staff determined that the physical phenomena modeled and described in the theory manual are adequate to calculate the accident scenario considered. The NRC staff concludes that this criterion is satisfied.

A.4.2.3.3 Field Equations

Field Equations

The field equations of the evaluation model should be adequate to describe the set of physical phenomena that occur in the accident.

SRP Section 15.0.2, Subsection III.3b

The licensee provided justification that the field equations were accurate in response to SNPB-3-12 (Reference 16). In that response, the licensee provided a description of the field equations being used. The licensee's EM employs a two-fluid model for two-phase flow with seven total field equations (mass, momentum, energy, and mixture energy), in one-dimensional form. Thus, while the computer code is called RELAP5-3D, only 1D components were used in the LTCC EM.

Because the licensee used an industry-standard two-fluid model which can account for non-equilibrium effects between the vapor and liquid phases and the standard 1D implementation of the field equations, the NRC staff determined that the field equations adequately model the physical phenomena of interest. The NRC staff concludes that this criterion is satisfied.

A.4.2.3.4 Validation of the Closure Relationships

Validation of the Closure Relationships

The range of validity of the closure relationships should be specified and should be adequate to cover the range of conditions encountered in the accident scenario.

SRP Section 15.0.2, Subsection III.3b

The licensee provided information on the validation of the closure relationships in response to SNPB-3-13 (References 16 and 19). In the initial response, the licensee provided a map from the phenomena modeled during a LOCA to the validation of those phenomena in the RELAP5-3D manual. In the supplemental response, the licensee discussed the validation of the key closure relationships for the LTCC EM and separated the closure relationships into five main areas: (1) flow regime maps, (2) energy closure relations, (3) momentum closure relations, (4) flow process models, and (5) other models (e.g., models for special components, reactor kinetics).

As discussed in Section A.3.1, "NRC Staff Evaluation Method," of this In-Vessel Thermal-Hydraulic Analysis, the NRC staff did not consider every closure model used in the LTCC EM, but instead focused on two sets of relationships: (a) those closure relationships of key physical phenomena for phase 4 (identified in Section A.4.2.1.3, "Phenomena Identification and Ranking"), and (b) all other closure relationships used in the LTCC EM. These are discussed below.

<u>Key Closure Relationships</u>

There were two highly ranked phenomena identified in phase 4 of the LTCC EM: natural convection heat transfer and counter current flow limitation (CCFL). For the first highly ranked phenomenon, the licensee stated that the heat transfer in the natural convection flow regime was modeled using the Chen correlation for both saturated and subcooled nucleate boiling. Because this correlation was used previously and is commonly used to predict such heat transfer, the NRC staff determined that this closure model generates appropriate predictions of the underlying phenomena.

For the second highly ranked phenomenon, the licensee stated that the counter current flow limitation was modeled using the Wallis CCFL correlation with smooth edges. The CCFL model is applied at the top of the core. The licensee performed a sensitivity study comparing three different CCFL models which could be used at this location. The data used to assess the various CCFL models was obtained from multiple tests (Reference 31). In general, a test was defined as two spaces separated by a horizontal plate which contained holes. The tests were characterized by the plate thicknesses, the various diameters of the holes, the number of holes, and pitch between the holes. When comparing the superficial velocities of gas and liquid, the number of holes seemed to be the largest predictor of the data's behavior, as more holes would support higher superficial velocities and hence better mixing.

Each of the three correlations studied—Wallis with sharp edges, Wallis with smooth edges, and Bankoff—was well correlated to a different data set (i.e., group of data with similar corresponding number of holes). The Wallis correlation with sharp edges conservatively predicted all the data (i.e., calculated a minimum of superficial velocities and therefore reduced mixing). While this correlation seemed appropriate in situations in which there were few holes, it greatly under-predicted the superficial velocities in situations where there were many holes. Wallis with smooth edges over-predicted the superficial velocities in data with a few number of holes, but under-predicted the superficial velocities in data with a moderate number of holes and greatly under-predicted the superficial velocities of data with a large number of holes. This was the correlation chosen by the licensee for the LTCC EM.

While Bankoff provided the best prediction of the data with a large number of holes, it significantly over-predicted the superficial velocities from the data with a moderate and a small number of holes. Thus, though Bankoff was likely the most realistic choice for the number of holes that exist in the upper core plate, the licensee chose to use the more conservative Wallis with smooth edges CCFL correlation. To demonstrate the impacts of the CCFL model, the licensee also performed a sensitivity study where each of the three models was used for a simulation. The Wallis/sharp model had the least mixing between the core and the upper plenum, and this resulted in core uncovery which occurred at various times after full-core blockage. The Wallis/smooth model did not show core uncovery, but resulted in fluctuations of the PCT, likely due to fluctuations of pressure in the core caused by void formation. The Bankoff model resulted in no such fluctuations. This was likely due to the high mixing predicted by this model as void formation in the core would not prevent liquid from reaching the core from the upper plenum, and therefore the PCT remained relatively constant and consistent. Because the Wallis with smooth edges was demonstrated to result in an accurate prediction of the relevant test data, and would likely result in an over-prediction of conditions causing CCFL which would reduce the mixing to the core, the NRC staff determined that this closure model will generate appropriate predictions of the underlying phenomena.

The NRC staff also notes that the meshing of the core alone is likely a significant conservatism. In the analysis, there is little time in the long-term phase when there is not water in the upper plenum, and therefore the only impedance to cooling the core is the CCFL which could occur on the upper core plate. It is likely that if CCFL could be completely excluded, then the entire simulation could be simplified to ensure that any delay in the SI flow reaching the core following full-core blockage would not result in core uncovery. However, the potential for CCFL does add a significant complication in that there can be adequate cooling flow in the upper plenum, but that flow may not be able to penetrate through the upper core plate and into the core itself. However, the NRC staff notes that the core noding used is likely to exaggerate the impacts of CCFL. In the simulations, the core itself is modeled using multiple axial nodes but a single radial node and the upper core plate is a single node on top of the core. Thus, any disturbance in the flow between the core and the upper plenum is experienced over the entire node. In reality, while CCFL may occur in hotter channels which generate significant amounts of steam (e.g., those commonly found in central regions of the core), it would not be likely to occur in lower power channels (e.g., those commonly found in the core periphery). Therefore, a likely SI flow path is down through the core support plate on the periphery, down the periphery fuel bundles, and then into the center of the core to make up for any loss due to boil off. Given the open lattice nature of the core, this flow path would in reality be almost unaffected by CCFL. However, due to the manner in which the core is meshed and the CCFL model is applied, this flow path is not possible in the simulation. The NRC staff finds that this is likely to be a conservatism in the licensee's analysis.

<u>Other closure relationships</u>

The closure relationships for the LTCC EM are found in Chapter 4 of the RELAP5-3D manual. The NRC staff reviewed the appropriate closure relationships and found that many of the closure models are commonly used to model phenomena under similar conditions as those found in the LTCC EM. The NRC staff determined that the other closure models will generate appropriate predictions of the underlying phenomena because these closure models are mature, the RELAP5 series of codes has wide use, and the phenomena observed during most of the simulations is well known and consistent with expectations (i.e., appropriate behavior during blowdown, refill, reflood).

Based on its review, the NRC staff determined that the range of validity of the closure relationships have been specified and are adequate to predict their underlying phenomena, because the key phenomena are relatively simple in nature, and because of likely conservatisms in the treatment of both core radial noding and CCFL modeling. Thus, the NRC staff determined that the closure relationships are adequate to cover the range of condition encountered in the accident scenario; therefore, this criterion is satisfied.

A.4.2.3.5 Simplifying and Averaging Assumptions

Simplifying and Averaging Assumptions

The simplifying assumptions and assumptions used in the averaging procedure should be valid for the accident scenario under consideration.

SRP Section 15.0.2, Subsection III.3b

The licensee provided justification that the simplifying and averaging assumptions were accurate in response to SNPB-3-14 (Reference 16). In the response, the licensee referenced the theory manual for RELAP5-3D (Reference 9) and provided a brief summary of the field equations used in the RELAP5-3D analysis. They also provided a brief description of how those equations were solved and a reference describing how they were derived.

The NRC staff reviewed the licensee's responses and determined that the simplifying assumptions used in the averaging procedure are valid for the accident scenario because the LTCC EM uses a well-documented and commonly used approach in averaging, and the scenarios under consideration do not contain phenomena that would challenge the averaging approach used in RELAP5-3D. Thus, the NRC staff concludes that this criterion is satisfied.

A.4.2.3.6 Level of Detail in the Model

Level of Detail in the Model

The level of detail in the model should be equivalent to or greater than the level of detail required to specify the answer to the problem of interest.

SRP Section 15.0.2, Subsection III.3b

The licensee provided justification that the level of detail in the models were accurate to simulate the problem of interest in response to SNPB-3-15 (Reference 18). In that response, the licensee provided a summary of the level of detail included in the LTCC EM and confirmed that the current modeling is consistent with STP's licensing basis analysis.

The NRC staff reviewed the licensee's responses and determined that STPNOC used an appropriate level of detail in the LTCC EM because the overall approach STPNOC described is consistent with that of a typical LOCA analysis; the phenomena of interest which most influence the figure of merit are known, well studied, and commonly modeled phenomena; and STPNOC performed appropriate sensitivity studies confirming the behavior of the LTCC EM (see Section A.4.2.4.7, "Sensitivity Studies"). Thus, the NRC staff concludes that this criterion is satisfied.

A.4.2.3.7 Equations and Derivations

Equations and Derivations

The equations and derivations should be correct.

SRP Section 15.0.2, Subsection III.3b

The manual for the RELAP5-3D code does not provide the specific derivation of equations, but instead relies on references for the development of such equations, since those derivations have been previously performed. The NRC staff determined that the equations and derivations are correct because these equations were previously derived and recorded in numerous references, and the key phenomena are well understood. Thus, the NRC staff concludes that this criterion is satisfied.

A.4.2.3.8 Similarity and Scaling

Similarity and Scaling

The similarity criteria and scaling rationales should be based on the important phenomena and processes identified by the accident scenario identification process and appropriate scaling analyses. Scaling analyses should be conducted to ensure that the data and the models will be applicable to the full scale analysis of the plant transient.

SRP Section 15.0.2, Subsection III.3b

In the analysis provided by the licensee supporting the modeling choices employed in the LTCC EM, one set of assessment data was used. The NRC staff reviewed the assessment data and determined that use of this data would be applicable to the full scale analysis of the plant transient because the assessment data was obtained over a wide range of conditions, and the data which was most applicable to reactor scale was conservatively predicted (as detailed in the discussion of the CCFL correlation in Section A.4.2.3.4, "Validation of the Closure Relationships"). Thus, the NRC staff concludes that this criterion is satisfied.

A.4.2.4 Code Assessment

The code assessment considers all code models against applicable experimental data and/or exact solutions in order to demonstrate that the code is adequate for analyzing the chosen scenario. The focus in Section A.4.2.3, "Evaluation Model," above, was on the field equations, the closure relationships, and the phenomena they model. The field equations and closure relationships were considered separately. In this section, the focus shifts to the combined use of all field equations and closure relationships in generating the figures of merit.

Section 15.0.2, Subsection III.3.d of the SRP contains eight review criteria for the code assessment. The review criteria topics and the sections providing the NRC staff's review are listed in Table 8.

Table 8: Code Assessment Review Categories

Subsection	
A.4.2.4.1	Single Version of the Evaluation Model
A.4.2.4.2	Validation of the Evaluation Model
A.4.2.4.3	Range of Assessment
A.4.2.4.4	Numerical Solution
A.4.2.4.5	Code Tuning
A.4.2.4.6	Compensating Errors
A.4.2.4.7	Sensitivity Studies
A.4.2.4.8	Assessment Data

A.4.2.4.1 Single Version of the Evaluation Model

Single Version of the Evaluation Model

All assessment cases should be performed with a single version of the evaluation model.

SRP Section 15.0.2, Subsection III.3d

The licensee stated that a single version of the EM was used for the submitted analysis in response to SNPB-3-16 (Reference 16). The NRC staff concludes that this criterion is satisfied because the licensee confirmed that a single version of the LTCC EM was used for the LTCC analysis.

A.4.2.4.2 Validation of the Evaluation Model

> **Validation of the Evaluation Model**
>
> *Integral test assessments must properly validate the predictions of the evaluation model for the full size plant accident scenarios. This validation should cover all of the important code models and the full range of conditions encountered in the accident scenarios.*
>
> SRP Section 15.0.2, Subsection III.3d

The licensee provided information on the validation of the EM in response to SNPB-3-17 (References 18 and 19). In that response, STPNOC provided a summary of the verification of the input models, and a summary of their judgment of the simulation results. The licensee also provided a discussion of the results of the simulations performed with the LTCC EM and how those results are consistent with the expected behavior following a LOCA. However, in general, there are no readily available test data which can be used to provide validation for the LTCC EM. Due to the lack of test data, the NRC staff used engineering judgment to conclude that the EM would produce an adequate prediction of the underlying phenomena. The staff based this conclusion on the following four considerations.

First, the NRC staff found that the analysis performed by the licensee contains a large number of conservatisms, including the following licensee assumptions:

- imposing a full-core blockage once the RCS system contains debris at 15 gm/FA, when the core would not be expected to fully block until some higher value of debris (as observed in testing supporting WCAP-16793) in the SE
- ignoring flow through the barrel-baffle region, even though test data demonstrates it remains unblocked
- ignoring flow through the holes between the barrel-baffle region and the core, even though these holes would likely remain unblocked
- biasing key input parameters conservatively including RWST volume, RWST temperature, and ECCS cooling
- using a conservative CCFL model and core modeling which is likely to exaggerate the impact of CCFL and underestimate the true mixing in the core

Second, the NRC staff found that even with these conservative assumptions, the resulting simulation of the long-term cooling phase was relatively simple. The majority of the SI flow was forced to flow through the upper plenum before it could flow out the break. There were few complex phenomena, and the only real complexity was caused by the potential for CCFL, which reduced the flow of the water from the upper plenum through the upper core plate and into the core.

Third, the NRC staff observed test data which demonstrated that the barrel-baffle region would remain open and would not be blocked by debris. A sensitivity study performed by the licensee demonstrated that with the barrel-baffle region unblocked, the core would not experience uncovery. A sensitivity study demonstrated that relaxing this conservatism, while keeping the others, reduced the complexity of the simulation to a simple mass balance equation because an alternative flow path was available for the SI flow to reach the core without being impacted by CCFL. It should be noted that this sensitivity only allowed flow through the barrel-baffle region. It did not allow flow through the core baffle holes, which would have further increased the coolant flow to the core.

Fourth, a sensitivity study by the licensee suggested that even if the barrel-baffle region could remained blocked, the core would not necessarily experience uncovery, provided additional radial channels were added to the core noding in the EM. While CCFL was found to restrict the flow from the upper plenum into the core in the analysis, it is possible that this results from an oversensitivity to CCFL caused by modeling the core – specifically, the top of the core where CCFL occurs – as a single radial node. Given the large flow area at the top of the core, the NRC staff finds it reasonable that CCFL would occur above the relatively hotter channels but likely not above the colder channels (e.g., the periphery). Above the colder channels, water from the upper plenum would flow down through the upper core plate and into the fuel bundles, where it would flow into the hotter portions of the core at elevations where CCFL would not be a consideration.

Based on the four conservatisms discussed above, the NRC staff concludes that the LTCC EM produced an accurate or conservative simulation and, therefore, this criterion is satisfied.

A.4.2.4.3 Range of Assessment

Range of Assessment

All code closure relationships based in part on experimental data or more detailed calculations should be assessed over the full range of conditions encountered in the accident scenario by means of comparison to separate effects test data.

SRP Section 15.0.2, Subsection III.3d

The issue of range of conditions was considered in the NRC staff's assessment of the validation of the closure relationships (see Section A.4.2.3.4, "Validation of the Closure Relationships") and the integral tests (see Section A.4.2.4.2, "Validation of the Evaluation Model"). Based on the conclusions in Sections A.4.2.3.4 and A.4.2.4.2, the NRC staff concludes that code closure relationships were assessed over the range of conditions encountered in the accident scenario, compared to separate test data, and found to be acceptable; therefore, this criterion is satisfied.

A.4.2.4.4 Numerical Solution

Numerical Solution

The numerical solution should conserve all important quantities.

SRP Section 15.0.2, Subsection III.3d

The NRC staff confirmed that the important quantities in mass, momentum, and energy are directly modeled. Further, the NRC staff confirmed that the QAP under which the simulations were performed, directed analysts to ensure proper convergence for each run (see the response to SNPB-3-25 in Reference 18).

The NRC staff determined that the numerical solution will conserve all important quantities such that the figures of merit can be adequately predicted because of the field equations (i.e., conservation equations) used by the licensee and the QAP direction to ensure proper convergence. Thus, the NRC staff concludes that this criterion is satisfied.

A.4.2.4.5 Code Tuning

Code Tuning
All code options that are to be used in the accident simulation should be appropriate and should not be used merely for code tuning.
SRP Section 15.0.2, Subsection III.3d

During the NRC staff's review of the LTCC EM, the NRC staff found the code options chosen to be appropriate and did not find any evidence of code tuning. Additionally, the sensitivity studies evaluated in Section A.4.2.4.7, "Sensitivity Studies," provide further evidence of the appropriateness of the code option choices.

Because the code options were appropriate for the scenarios simulated and the sensitivity studies demonstrated robustness in key aspects of performing the simulations, the NRC staff determined that the LTCC EM was not artificially tuned. The NRC staff concludes that this criterion is satisfied.

A.4.2.4.6 Compensating Errors

Compensating Errors
The reviewers should ensure that the documentation contains comparisons of all important experimental measurements with the code predictions in order to expose possible cases of compensating errors.
SRP Section 15.0.2, Subsection III.3d

There were no direct comparisons of code predictions to experimental measurements. This issue is addressed fully in Section A.4.2.4.2, "Validation of the Evaluation Model." However, the NRC staff considered the potential for compensating error in the NRC staff's assessments of the initial and boundary conditions (see Section A.4.2.1.4, "Initial and Boundary Conditions"), the assessment of the validation of the closure relationships (see Section A.4.2.3.4, "Validation of the Closure Relationships"). The NRC staff also considered compensating errors due to experimental measurements and those that may have resulted from making the analysis "more conservative." Based on the NRC staff's review of the licensee's submittal and discussions in Sections A.4.2.1.4, A.4.2.3.4, and A.4.2.4.2, in this In-Vessel Thermal-Hydraulic Analysis, the NRC staff concludes that this criterion is satisfied.

A.4.2.4.7 Sensitivity Studies

Sensitivity Studies

Assessments should be performed where applicable [specific test cases for LOCA to meet the requirements of Appendix K to 10 CFR Part 50 and TMI [Three Mile Island] action items for PWR small-break LOCA}.

SRP Section 15.0.2, Subsection III.3d

Appropriate sensitivity studies shall be performed for each evaluation model, to evaluate the effect on the calculated results of variations in noding, phenomena assumed in the calculation to predominate, including pump operation or locking, and values of parameters over their applicable ranges. For items to which results are shown to be sensitive, the choices made shall be justified.

Appendix K to 10 CFR Part 50

A detailed analysis shall be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.

TMI [Three Mile Island] action items for PWR (Reference 28)

The NRC staff determined that the TMI action items are out of the scope of this review because the items were related to pressurized thermal shock, not PCT.

The licensee provided information on the following sensitivity studies in response to SNPB-3-18 (References 18 and 19) and SNPB-3-20 (References 18, 19, and 20). In those responses, the licensee provided details on the following sensitivity study topics:

- Core radial mesh sensitivity
- Core axial mesh sensitivity
- Appendix K decay heat with single worst failure and steam generator tube plugging
- Axial power shape
- Break size sensitivity
- Break orientation
- Open barrel-baffle region

<u>Core radial mesh sensitivity</u>

The licensee provided information on the core radial mesh sensitivity in response to SNPB-3-18 (References 18 and 19). In the sensitivity study, the licensee added a hot channel to the core region. This two-channel model was compared to the single core channel model (i.e., the base case). As expected, the conditions for CCFL occurred more frequently at the top of the hot channel than they did at the top of the average channel. This difference is greatest immediately after core blockage, when CCFL in the hot channel is nearly constant, while the average channel is not constant. Additionally, when the PCT is compared to the base case, both cases fluctuate in a similar manner near the saturation temperature, but the two-channel model is much smoother and at a slightly lower temperature than the base case. The NRC staff considers this likely because in the two-channel model, liquid can flow into the average channel and steam and liquid exit through the hot channel, but in the base case all liquid flowing into the core and all steam and liquid exiting the core must go through the same channel (i.e., the same node). The NRC staff finds that the licensee's treatment of the radial mesh of the core is acceptable because the single-channel model predicts conservatively compared to the simulated two-channel model, and because a model with a realistic number of channels would likely predict even better mixing between the core and upper plenum.

Core axial mesh sensitivity

The licensee provided information on the core axial mesh sensitivity in response to SNPB-3-18 (Reference 19). In this sensitivity study, the licensee reduced the axial mesh in the core from 21 axial nodes to 10 axial nodes. The timings of major events during the simulations and the resulting PCT from both cases were nearly identical. Because the sensitivity demonstrates that a change in axial mesh size does not impact the simulations results and because the licensee is using an axial mesh size similar to that used for LOCA analysis, the NRC staff finds that this treatment of the axial mesh of the core is acceptable.

Appendix K decay heat with single worst failure and steam generator tube plugging

The licensee provided a sensitivity study discussing the impacts of assuming Appendix K decay heat, single worst failure, and steam generator tube plugging in response to SNPB-3-20 (References 18, 19, and 20). This study included Appendix K decay heat (i.e., a 1.2 multiplier on the 1971 American Nuclear Society decay heat standard), the failure of a single train of ECCS, and steam generator tube plugging of 10 percent. In this study, the licensee did not assume some of the conservatisms found in the LTCC EM; namely, the licensee assumed a nominal RWST volume instead of the lower volume used in the base case, and nominal RWST and sump pool temperatures instead of the conservatively high temperatures used in the base case. A comparison to the base case showed that the two cases have very similar timings up through the reflood phase. However, sump switchover was delayed in the Appendix K sensitivity due both to the larger RWST volume and the reduced ECCS flow resulting from the assumed failure of a single train of ECCS. In both the base case and the Appendix K sensitivity study, the PCT and the final pressures are similar, and the temperature of the cladding is approximately the coolant's saturation temperature.

The NRC staff requested the Appendix K decay heat sensitivity study to determine the impacts of assuming Appendix K decay heat and single worst failure on the event. The staff reviewed the licensee's results and determined that a failed train of ECCS would delay the drainage of the RWST, resulting in delayed sump switchover/core blockage, and therefore a lower core power when blockage occurs. However, it is unclear from the licensee's response (1) how sensitive the timing of blockage is to the assumed decay heat level, (2) how much of an impact the failed ECCS train has on the blockage timing, and (3) how much of an impact the RWST volume has on the timing. While these items were not addressed in the licensee's sensitivity studies, the NRC staff finds that any increase in decay heat would have a minimal impact on the PCT, since the core would simply stabilize in the long-term phase at a slightly higher pressure and, therefore, slightly higher saturation temperature. As discussed above, the NRC staff finds the treatment of the 10 CFR 50 Appendix K decay heat sensitivity study is acceptable.

Axial power shape

The licensee provided information on the axial power shape in response to SNPB-3-20 (References 18, 19, and 20). This sensitivity study is evaluated in the discussion on core uncovery in Section A.4.2.1.3, "Phenomena Identification and Ranking."

Break size sensitivity

The licensee provided information on the break size sensitivity in response to SNPB-3-2 (Reference 19). This sensitivity study is evaluated in the discussion on bounding break size assumptions in Section A.4.2.1.4, "Initial and Boundary Conditions."

Break Orientation

The licensee provided information on the break size sensitivity in response to SNPB-3-20 (Reference 20). In its response, the licensee stated that the results of the study indicated that the break orientation had no significant impact on the resulting PCT. Because the break orientation had no impact on the results of the PCT, the NRC staff finds that the treatment of the break orientation is appropriate.

Open barrel-baffle bypass region

The licensee provided information on the barrel-baffle bypass sensitivity study in response to SNPB-3-20 (References 18, 19, and 20). The licensee also re-performed this sensitivity study assuming the worst power shape (top-skewed). As discussed in Section A.4.2.1.3, "Phenomena Identification and Ranking," the NRC staff determined that this sensitivity is likely to be the most realistic of all of the analysis performed, and demonstrates that there is no core uncovery during LTCC.

The NRC staff finds that appropriate test cases were performed because the comparisons demonstrated that even under-conservative conditions, the core temperature remains below the 800 °F acceptance criterion, thus, under more realistic but still conservative conditions, the core will likely not experience uncovery and will avoid exceeding the saturation temperature. Therefore, the NRC concludes that this criterion is satisfied.

A.4.2.4.8 Assessment Data

Assessment Data

Published literature should be referred to for sources of assessment data for specific phenomena, accident scenarios, and plant types.

SRP Section 15.0.2, Subsection III.3d

One set of assessment data were used in the licensee's analysis supporting the LTCC EM. Because the assessment data was obtained from published literature (see Reference 31) containing references to multiple other well-known publications, the NRC staff found this use of assessment data appropriate. The NRC staff concludes that this criterion is satisfied.

A.4.2.5 Uncertainty Analysis

Uncertainty analyses are performed to confirm that the combined code and application uncertainty is less than the design margin for the safety parameter of interest when the code is used in a licensing calculation. Safety parameters are those parameters which have limits to ensure plant safety, such as the specified acceptable fuel design limits required by General Design Criterion 10, "Reactor design," in Appendix A to 10 CFR Part 50. Examples of safety parameters are PCT, cladding oxidation thickness, departure from nucleate boiling ratio, and critical power ratio.

No explicit uncertainty analysis was prescribed or performed for the LTCC EM. However, the NRC staff reviewed specific aspects of the LTCC EM to confirm that specific uncertainties would be accounted for in the analysis.

Section 15.0.2, Subsection III.3.e of the SRP contains three criteria for the uncertainty analysis. The review criteria topics and the subsections providing the NRC staff's review are listed in Table 9.

Table 9: Uncertainty Analysis Review Categories

Subsection	
A.4.2.5.1	Important Sources of Uncertainty
A.4.2.5.2	Experimental Uncertainty
A.4.2.5.3	Calculated and Predicted Results

A.4.2.5.1 Important Sources of Uncertainty

Important Sources of Uncertainty

The accident scenario identification process should be used in identifying the important sources of uncertainty. Sources of calculation uncertainties should be addressed, including uncertainties in plant model input parameters for plant operating conditions (e.g., accident initial conditions, set points, and boundary conditions). To address these uncertainties, demonstrate that the combined code and application uncertainty should be less than the design margin for the safety parameter of interest in the calculation.

SRP Section 15.0.2, Subsection III.3e

The licensee provided justification that important sources of uncertainty were identified in responses to RAI questions. First, the licensee identified and provided a description of the important uncertainties in response to SNPB-3-32 (Reference 18). Second, the licensee provided a discussion of the impact of the important uncertainties on the analysis in response to SNPB-3-22 (Reference 18). Third, the licensee described how the uncertainties were accounted for in the input in response to SNPB-3-21 (Reference 18). Finally, the licensee provided a discussion of how the important uncertainties were addressed in the input deck for the LTCC EM in response to SNPB-3-23 (Reference 18).

The following is a listing of the uncertainties identified by STPNOC, a summary of how those uncertainties were addressed, and the NRC staff's review each uncertainty.

<u>Initial Reactor Power</u>

The licensee used nominal reactor power (i.e., 3,853 megawatt-thermal) for the LTCC EM. The NRC staff determined previously in this In-Vessel Thermal-Hydraulic Analysis that the reactor power will have a minimal impact during the long-term phase and finds this treatment is acceptable. It should be noted that the licensee performed a sensitivity study on the decay heat used in the analysis, which is addressed in Section A.4.2.4.7, "Sensitivity Studies." This study demonstrated that a large change in decay heat (which is equivalent, for the figure of merit, to a change in the initial reactor power) may result in a small change in PCT. However, this change in PCT is due to the change in the saturation temperature. Because the PCT remains linked to the saturation temperature and well below the limit of 800 °F, the NRC staff finds that the licensee's treatment of the uncertainty is acceptable and, therefore, the initial reactor power uncertainty is acceptable.

Core Heat Structure Thermal Properties

No conservatisms were added to better address the core heat structure thermal properties, as the licensee stated these properties are of low significance during the long-term cooling phase. The NRC staff concluded that these properties would have a minimal impact on the PCT during the long-term phase and, at most could very slightly increase the heat transferred to the fluid during the long-term phase, since there is ample water to remove the heat during that phase, and any increase would be small and inconsequential. Thus, the NRC staff finds that no additional conservatism is needed and the licensee's treatment of the core heat structure thermal properties is acceptable; therefore, the core heat structure thermal properties uncertainty is acceptable.

Reactor Vessel Passive Heat Structures

The licensee did not add conservatism to better address the reactor vessel heat structures. However, the licensee included the metal mass from these structures in the LTCC EM and calculated the impact of the stored energy on the fluid in the RCS. The NRC staff notes that during the long-term cooling phase, most of the passive heat structures have been in contact with coolant for a substantial portion of the accident and have had ample opportunity to lose stored energy. Typically, the staff would expect to see the heat transferred from the steam generators to the primary coolant dominate over any remaining heat transferred from other portions of the RCS. The licensee performed a sensitivity study (discussed in Section A.4.2.1.3, "Phenomena Identification and Ranking") which demonstrated that the heat transferred from the steam generators had a minimal impact on the conditions in the core. The NRC staff finds that the licensee accounted for the passive heat structures in its LTCC EM and performed a sensitivity study which demonstrates that the remaining passive heat has little impact on conditions in the core. Therefore, the licensee's treatment of the uncertainty associated with the reactor vessel passive heat structures is acceptable and, therefore, the uncertainty is acceptable.

Reactor Core Axial Power Shape

The licensee performed a sensitivity study on the axial power shape. That sensitivity study and the use of the cosine axial power shape is evaluated in Section A.4.2.4.7, "Sensitivity Studies." The NRC staff found that the while the axial power shape had an impact on the given base case, that impact was due largely to other conservative assumptions, such as a blocked barrel-baffle region or the choice of modeling the core using a single radial node. When those assumptions were relaxed to result in a more realistic analysis, the sensitivity to the axial power shape was minimized. The NRC staff finds that the licensee appropriately accounted for the uncertainty associated with the reactor core axial power shape in its LTCC EM and, therefore, the reactor core axial power shape is acceptable.

Steam Generator Tube Plugging

The licensee performed a sensitivity study on steam generator tube plugging. However, in that sensitivity study, a number of additional parameters were varied simultaneously which were expected to have a much larger impact on the results than the tube plugging. In general, the NRC staff does not consider tube plugging to be an important uncertainty to capture, as its impact would likely be small. Further, it was not clear to the NRC staff if more plugging or less plugging would be conservative. More plugging increases the pressure drop through a steam generator, but removing tubes from service also means that less water is needed to fill the steam generator. The licensee chose to assume 0 percent tube plugging in the main analysis. Given that this would require the maximum amount of water to fill up the steam generators, which would cause the largest delay between full-core blockages and when SI flow reaches the core, the NRC staff finds the licensee's treatment of the uncertainty associated with steam generator tube plugging conservative and, therefore, the uncertainty is acceptable.

Vessel Flow Bypass Fractions

The licensee identified six bypass flows in response to SNPB-3-3 (Reference 16). Those six bypass flows are as follows:

1. Thimble tube flow through the fuel rods
2. Core former-to-fuel gap flows
3. LOCA holes flow between the barrel-baffle region and the core
4. Barrel-baffle flow between the bottom of the core and the top of the core
5. Cold-leg to hot-leg leakage flow
6. Upper head spray nozzle flow

The licensee conservatively chose to keep just one of the six bypass flows unblocked after full-core blockage: the upper head spray nozzle flow. The nodalization of this portion of the reactor vessel in the EM is discussed below. The licensee's analysis does not predict that it is an important flow path because the primary path for SI following core blockage is through the steam generators. The NRC staff finds the licensee's treatment of the uncertainty in the vessel flow bypass fractions is acceptable because of its conservative treatment (i.e., disregarding and minimizing) of other bypass flow paths which could provide substantial cooling to the core. Therefore, the vessel flow bypass fractions uncertainty is acceptable.

Core Nodalization

The licensee provided a sensitivity study of the core nodalization in response to SNPB-3-18 (References 18 and 19). That sensitivity study is evaluated in Section A.4.2.4.7, "Sensitivity Studies." Because the core nodalization used by STPNOC is very similar to the core nodalization commonly used for LOCA analysis by the industry, and because the sensitivity study verified the relative insensitivity of the PCT during the long-term phase to the core nodalization, the NRC staff finds that the licensee's treatment of the uncertainty in the core nodalization is acceptable. Therefore, the core nodalization uncertainty is acceptable.

Upper Head Nodalization

The licensee provided a description of the upper head nodalization in response to SNPB-3-22 (Reference 18). Early in the review process, the dominant SI flow path was into the cold-leg, into the top of the downcomer, through the upper head spray nozzles, into the upper head, down through the control rod guide tubes and into the top of the core. The licensee modified the nodalization of the upper head to better account for the standpipe effect of the control rod guide tubes. However, in the final scenarios considered by the licensee, this cooling path became inconsequential compared to the SI flow from the steam generator u-tubes. In light of the minimal impact of this flow path, the NRC staff finds that the new nodalization of the upper head adequately captures the standpipe effect from the guide tubes, and is therefore acceptable.

RWST Usable Volume

The licensee used an RWST volume which was conservatively biased lower than the expected usable volume, and even below the usable volume credited in a LOCA. Because this smaller volume is conservative for the analyses performed, the NRC staff finds this treatment of the RWST usable volume uncertainty is acceptable. Therefore, the RWST usable volume uncertainty is acceptable.

Decay Power Model

The licensee performed a sensitivity study on the core decay power. That sensitivity study and the use of the 10 CFR 50 Appendix K decay power is evaluated in Section A.4.2.4.7, "Sensitivity Studies." The NRC staff determined that any increase in decay heat would have a minimal impact on the PCT, since the core would be at a slightly higher pressure and, therefore, slightly higher saturation temperature. Thus, the NRC staff finds the licensee's treatment of the uncertainty associated with the decay power model is acceptable and, therefore, the decay power model uncertainty is acceptable.

Break Size

The licensee performed a sensitivity study on the various break sizes. That sensitivity study, and the use of the 16-inch break size, is evaluated in Section A.4.2.1.4, "Initial and Boundary Conditions." The NRC staff determined the 16-inch break most limiting compared to smaller breaks. The NRC staff finds that the licensee's treatment of the uncertainty associated with breaks size to be acceptable and, therefore, the break size uncertainty is acceptable.

Break Orientation

The licensee performed a sensitivity study on the break orientation. That sensitivity study and the use of the orientation chosen is evaluated in Section A.4.2.4.7, "Sensitivity Studies." The NRC staff determined that the break orientation has no impact on the PCT during LTCC. Thus, the licensee's treatment of the uncertainty in break orientation is acceptable and, therefore, the break orientation uncertainty is acceptable.

ECCS Flow Rate

The licensee did not consider a single worst failure of an SI train in its LTCC EM. This is conservative with respect to refill and reflood PCT since the failure means it takes longer to quench the core and PCT will be higher. However, it is non-conservative with respect to LTCC PCT since the extra train of SI causes the RWST to drain faster, which in turn causes complete core blockage sooner and at a higher decay heat power. Because the licensee treated the ECCS flow rate in a manner which would result in a higher than PCT than would occur if the failure of an ECCS train were assumed, the NRC staff finds this treatment of the uncertainty in the ECCS flow rate conservative and, therefore, the ECCS flowrate uncertainty is acceptable.

ECCS Injection Temperature

The licensee used higher than expected ECCS temperatures during both the SI phase (i.e., phases 1 and 2) and the recirculation phase (i.e., phases 3 and 4). The ECCS injection temperatures are evaluated in Section A.4.2.1.4, "Initial and Boundary Conditions." The NRC found that the licensee's use of higher than expected temperatures for the ECCS injection was a conservative assumption. Therefore, the licensee's treatment of the uncertainty associated with ECCS injection temperature is acceptable and, therefore, the uncertainty is acceptable.

Core Barrel-Baffle Bypass Fraction

The licensee modeled the core barrel-baffle bypass as fully blocked which is a conservative assumption such that no cooling can reach the core through this flowpath. Because this assumption is conservative (for more details, see Section A.4.2.4.7, "Sensitivity Studies"), the NRC staff determined that the licensee's treatment of the uncertainty associated with the core barrel-baffle bypass fraction is acceptable and, therefore, the uncertainty is acceptable.

Core Blockage Fraction

The licensee initiated blockage at the time of sump switchover, and assumed 100 percent blockage 360 seconds later. Because 360 seconds is a conservatively early estimate of the time it would take to completely block the core and because assuming the entire core is blocked is a conservative assumption, as it reduces the ability for coolant to flow into the core, the NRC staff found that the licensee's treatment of the uncertainty associated with core blockage fraction and time to core blockage is acceptable; therefore, the uncertainty is acceptable.

Plant Set Points and Delays

The licensee used nominal values (i.e., the values given in the technical specifications) for the various plant set points and delays. Because these values are consistent with the technical specification values, the NRC staff found that the licensee's treatment of the uncertainty associated with plant set points and delays is acceptable; therefore, the uncertainty is acceptable.

CCFL Parameters

The CCFL parameters were evaluated in in Section A.4.2.3.4, "Validation of the Closure Relationships." The NRC staff determined that the CCFL model chosen conservatively predicts CCFL by reducing the mixing between the core and the upper plenum below what would be reasonably expected. Therefore, the NRC staff finds the uncertainty associated with CCFL parameters is acceptable.

Because the licensee addressed the key sources of uncertainty as discussed above, and because with these uncertainties addressed, there is significant margin to the PCT limit, the NRC staff determined that important sources of uncertainty are identified and accounted for in an appropriate manner. These analyses demonstrate that the combined code and application uncertainty would be less than the design margin for the PCT (i.e., there is ample margin to the actual PCT limit). The NRC staff concludes that this criterion is satisfied.

A.4.2.5.2 Experimental Uncertainty

Experimental Uncertainty
The uncertainties in the experimental data base should be addressed. Data sets and correlations with experimental uncertainties that are too large when compared to the requirements for evaluation model assessment should not be used.
SRP Section 15.0.2, Subsection III.3e

The licensee did not perform a direct assessment of experimental uncertainty during its review since it is incumbent on the NRC staff to perform this assessment. The NRC staff considered the results of a number of experiments from a published document (Reference 31) to address uncertainty in support of the CCFL closure model. For the CCFL study reviewed, the NRC staff determined that since the experiments conducted by different researchers at different facilities and times were found to be repeatable and similar (approximately 10 percent difference), then there is no need to further evaluate the experimental uncertainties of the provided data. The NRC staff concludes that this criterion is satisfied.

A.4.2.5.3 Calculated and Predicted Results

Calculated and Predicted Results
For separate effects tests and integral effects tests, the differences between calculated results and experimental data for important phenomena should be quantified for bias and deviation.
SRP Section 15.0.2, Subsection III.3e

Test data is not available to provide validation for the LTCC EM so the NRC staff used engineering judgment to conclude that the EM would produce an adequate prediction of the underlying phenomena. This is the specific focus of Section A.4.2.4.2, "Validation of the Evaluation Model," and, holistically, of the LTCC EM safety evaluation.

For CCFL, the review was more a review of what coefficients were most applicable based on existing experiments, than a review of the experimental data itself. As no separate effects or integral effects tests were used to support the validation of the LTCC EM, the NRC staff concludes that this criterion does not apply to this review.

A.4.2.6 Quality Assurance Program

The QAP covers, in part, the procedures for design control, document control, software configuration control and testing, and error identification and corrective actions used in the development and maintenance of the LTCC EM. The QAP also ensures adequate training of personnel involved with code development and maintenance, as well as those who perform the analyses.

Section 15.0.2, Subsection III.3.f of the SRP contains three review criteria for the QAP. The review criteria topics and the subsection providing the NRC staff's reviews are listed in Table 10.

Table 10: Quality Assurance Plan Review Categories

Subsection	
A.4.2.6.1	Appendix B Quality Assurance Program
A.4.2.6.2	Quality Assurance Documentation
A.4.2.6.3	Independent Peer Review

A.4.2.6.1 Appendix B Quality Assurance Program

Appendix B Quality Assurance Program

The evaluation model should be maintained under a quality assurance program that meets the requirements of Appendix B to 10 CFR Part 50.

SRP Section 15.0.2, Subsection III.3f

The licensee provided a discussion that the EM was maintained under a QAP that meets the requirements of 10 CFR 50 Appendix B in response to SNPB-3-19 (References 16 and 19) and SNPB-3-23 through SNPB-3-29 (Reference 18). In its response to SNPB-3-23, the licensee confirmed that the RELAP5-3D analysis performed for the LTCC EM was in compliance with the STP Appendix B QAP. In its response to SNPB-3-24, the licensee confirmed that the input values are compared against the reference source and are controlled using the QAP.

The licensee provided details on the correct installation and execution of RELAP5-3D in response to SNPB-3-19 (References 16 and 19). In that response, the licensee confirmed that there were no technical differences between the Idaho National Laboratory's results (i.e., the creators of RELAP5-3D) and the licensee's results for the given set of set cases.

The NRC staff noted that Volume 5, Section 2.2.4.2 of the RELAP5-3D theory manual (Reference 9) describes how RELAP5-3D results should be analyzed, and specifically discusses the following criteria:

- Code convergence
- Non-physical results
- Realistic results
- Boundary conditions are well behaved
- Thoroughly understood results

In its responses to SNPB-3-25 through SNPB-3-29 (Reference 18), the licensee provided a discussion of how the QAP would assure each of the above criteria for the RELAP5-3D results. In its response to SNPB-3-25, the licensee stated that proper convergence is often ensured by requiring a qualified analyst to perform the analysis, adding that the "mass error" is typically a good figure of merit to observe to ensure convergence. In response to SNPB-3-26 through SNPB-3-29, the licensee stated that ensuring that the results are physically appropriate and realistic, that the boundary conditions behave as prescribed, and that the results are thoroughly understood are also considerations of the analyst performing the analysis. The NRC staff considers that such reliance on qualified analysts to be common industry practice, and it is also one of the reasons why an independent peer review is part of the QAP. For the NRC's evaluation of the peer review, see Section A.4.2.6.3, "Independent Peer Review."

Initially, the NRC staff's RAI questions were written assuming that the LTCC EM would be used generically for future analyses. However, during the course of the review, the licensee changed its methodology and the NRC staff also changed its review strategy to limit the scope as discussed in Section A.3.2, "Scope of the Review." Future use of the LTCC EM would require additional review and acceptance by the NRC staff, as the NRC only reviewed the licensee's application of its QAP to the RELAP5-3D analyses as appropriate for this limited use by the licensee.

Because the licensee provided a process which addresses quality assurance and because the NRC staff verified aspects of the licensee's application of its QAP, the NRC staff finds that a simulation performed under the program would achieve a sufficient quality for use in reactor safety licensing analysis. The NRC staff concludes that this criterion is satisfied.

A.4.2.6.2 Quality Assurance Documentation

Quality Assurance Documentation

The quality assurance program documentation should include procedures that address all of these areas [design control, document control, software configuration control and testing, and corrective actions].

SRP Section 15.0.2, Subsection III.3f

The licensee provided justification that its QAP documentation includes procedures to address all relevant areas in response to SNPB-3-30 (Reference 18). In its response, the licensee referenced the response to SNPB-3-23, which stated that the QAP for the simulations requires compliance with all the elements of the STP 10 CFR 50 Appendix B program including: error reporting, qualification for personnel, software quality assurance, and records management.

Because the licensee is using a QAP consistent with its 10 CFR 50 Appendix B program, the NRC staff finds that the QAP documentation for RELAP5-3D includes procedures to address all relevant areas of interest. The NRC staff concludes that this criterion is satisfied.

A.4.2.6.3 Independent Peer Review

Independent Peer Review

Independent peer reviews should be performed at key steps in the evaluation model development process.

SRP Section 15.0.2, Subsection III.3f

The licensee provided justification that an independent peer review was performed at key steps in the execution of the LTCC EM in response to SNPB-3-31 (Reference 18). In its response, the licensee clarified that the QAP specified separate roles for the preparer and the checker in generating the simulations, both of whom must be fully qualified for performing the procedure in question.

The NRC staff finds that the QAP appropriately applied an independent peer review because the licensee confirmed that multiple layers of review were conducted by different individuals. Further, the licensee submitted the simulation results to the NRC staff for an additional independent peer review with acceptable results. The NRC staff concludes that this criterion is satisfied.

A.4.3 Conclusions

The NRC staff made the following conclusions based on the referenced evaluations provided in this In-Vessel Thermal-Hydraulic Analysis:

- Based on the staff's evaluation in Section A.4.2.1, "Accident Scenario Identification Process," the NRC staff finds that that the accident scenario identification process is a structured process and is appropriately used to identify the key figures of merit for the accident.

- Based on the staff's evaluation in Section A.4.2.2, "Documentation," the NRC staff finds that the documentation provided is sufficient to describe the LTCC EM.

- Based on the staff's evaluation in Section A.4.2.3, "Evaluation Model Development," the NRC staff finds that the individual field equations and closure relationships are adequate for modeling the phenomena determined to be important for the chosen scenario.

- Based on the staff's evaluation in Section A.4.2.4, "Code Assessment," the NRC staff finds that the code assessment demonstrates that the LTCC EM is adequate for analyzing the chosen scenario.

- Based on the staff's evaluation in Section A.4.2.5, "Uncertainty Analysis," the NRC staff finds that the uncertainties in the inputs and models are appropriately accounted for such that the results are expected to bound possible outcomes for the accident.

- Based on the staff's evaluation in Section A.4.2.6, "Quality Assurance Program," the NRC staff finds that the STPNOC QAP assures all relevant actions in the development and maintenance of the EM have been taken.

Based on the above, the NRC staff concludes that the STPNOC LTCC EM and the simulations performed specifically for STP, Units 1 and 2, provide a conservative analysis for debris impacts on LTCC for hot-leg breaks 16 inches in diameter and smaller. Further, the simulations performed with this EM along with those from LOCADM demonstrates that the stated acceptance criteria from WCAP-16793 have been satisfied:

- The EM used for the LTCC analysis demonstrates that the maximum clad temperature remains at the saturation temperature and therefore shall not exceed 800 °F following core quench and re-flooding.

- The LOCADM analysis demonstrates that the thickness of the cladding oxide and the deposits of material on the fuel shall not exceed 0.050 inches in any fuel region.

The NRC staff's conclusions in the LTCC EM are specific to STP and the analysis performed. Future use of this in-vessel thermal-hydraulic EM was not considered, because use of the EM outside the simulations reviewed could invalidate the NRC staff's conclusions. If the input and modeling assumptions are made less conservative, such as decreasing the severity of accident conditions (e.g., the amount of debris generated, post-LOCA PCT or oxide thickness), or if the relative importance of specific models or flow paths are changed, then the NRC staff's assessment is no longer applicable.

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A.5 REFERENCES

1. U.S. Nuclear Regulatory Commission, Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004 (ADAMS Accession No. ML042360586).

2. Nuclear Energy Institute, NEI 04-07, Volume 1, "Pressurized Water Reactor Sump Performance Evaluation Methodology," December 2004 (ADAMS Accession No. ML050550138).

3. Nuclear Energy Institute, NEI 04-07, Volume 2, "Safety Evaluation by the Office of Nuclear Reactor Regulation re to NRC Generic Letter 2004-02, Revision 0," dated December 6, 2004 (ADAMS Accession No. ML050550156).

4. PWR Owners Group, "Transmittal of WCAP-16793-NP-A, Revision 2, 'Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid,' dated July 2013," dated August 13, 2013 (ADAMS Package Accession No. ML13239A111).

5. Westinghouse Electric Company, LLC, WCAP-16793-NP-A, Revision 2, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," July 2013 (ADAMS Accession Nos. ML13239A114 (part 1 of 2) and ML13239A115 (part 2 of 2)).

6. U.S. Nuclear Regulatory Commission, "Revised Content Guide for Generic Letter 2004-02 Supplemental Responses," dated November 21, 2007 (ADAMS Package Accession No. ML073110389).

7. Ruland, W. H., U.S. Nuclear Regulatory Commission, letter to Anthony Pietrangelo, Nuclear Energy Institute, "Revised Content Guide for Generic Letter 2004-02 Supplemental Responses," dated November 21, 2007 (ADAMS Accession No. ML073110269).

8. Jordan, T.J., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "90-Day Response to Generic Letter 2004-02: Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," dated March 8, 2005 (ADAMS Accession No. ML050770105).

9. Idaho National Laboratory, "RELAP5-3D Code Manual: Volumes 1 – 5," INEEL-EXT-98-00834, Revision 4.2, July 2014.

10. Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Supplement 2 to STP Pilot Submittal and Requests for Exemptions and License Amendment for a Risk-Informed Approach to Address Generic Safety Issue (GSI)-191 and Respond to Generic Letter (GL) 2004-02," Attachments 1.1-1.4, dated August 20, 2015 (ADAMS Accession No. ML15246A127).

11. Regner, L., U.S. Nuclear Regulatory Commission, electronic mail to Wayne Harrison, STP Nuclear Operating Company, "STP GSI-191 DRAFT Questions," dated October 21, 2015 (ADAMS Accession No. ML16022A177).

12. Regner, L., U.S. Nuclear Regulatory Commission, electronic mail to Wayne Harrison, STP Nuclear Operating Company, "FW: STP Initial RAIs," dated December 11, 2015 (ADAMS Accession No. ML16022A176).

13. Regner, L., U.S. Nuclear Regulatory Commission, letter to Dennis L. Koehl, STP Nuclear Operating Company, "South Texas Project, Units 1 and 2- Request for Additional Information Related to Request for Exemptions and License Amendment for Use of a Risk-Informed Approach to Resolve the Issue of Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors," dated April 11, 2016 (ADAMS Accession No. ML16082A507).

14. Regner, L., U.S. Nuclear Regulatory Commission, letter to Dennis L. Koehl, STP Nuclear Operating Company, "South Texas Project, Units 1 and 2- November 17-19, 2015, Regulatory Audit associated with a Risk-Informed Solution to Generic Safety Issue 191 (CAC Nos. MF2400, MF2401, MF2402, MF2403, MF2404, MF2405, MF2406, MF2407, MF2408, and MF2409)," dated April 13, 2016 (ADAMS Accession No. ML16095A010)

15. Regner, L., U.S. Nuclear Regulatory Commission, letter to Dennis L. Koehl, STP Nuclear Operating Company, "South Texas Project, Units 1 and 2 - February 23-25, 2016, Regulatory Audit Summary for the Thermal-Hydraulic Review associated with a Risk-Informed Solution to Generic Safety Issue 191 (CAC Nos. MF2400 through MF2409)," dated May 11, 2016 (ADAMS Accession No. ML16127A400).

16. Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "First Set of Responses to April 11, 2016 Requests for Additional Information Regarding STP Risk-Informed GSI-191 Licensing Application," dated May 11, 2016 (ADAMS Accession No. ML16154A117).

17. Connolly, J., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Second Set of Responses to April 11, 2016 Requests for Additional Information Regarding Risk-Informed GSI-191 Licensing Application," dated June 16, 2016 (ADAMS Accession No. ML16196A241).

18. Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Third Set of Responses to April 11, 2016 Requests for Additional Information Regarding STP Risk-Informed GSI-191 Licensing Application, Response to SNPB RAIs," dated July 21, 2016 (ADAMS Accession No. ML16229A189).

19. Connolly, J., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Supplement 3 to Revised Pilot Submittal and Requests for Exemptions and License Amendment for a Risk-Informed Approach to Address Generic Safety Issue (GSI)-191 and Respond to Generic Letter (GL) 2004-02," dated October 20, 2016 (ADAMS Accession No. ML16302A015).

20. Connolly, J., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Additional Information Regarding Sensitivity Studies for STPNOC Risk-Informed Pilot GSI-191 Application," dated November 9, 2016 (ADAMS Accession No. ML16321A407).

21. Regner, L., U.S. Nuclear Regulatory Commission, letter to G. T. Powell, STP Nuclear Operating Company, "South Texas Project, Units 1 and 2 – Re: Closeout of Request for Additional Information Questions that are no Longer Applicable Associated with the Resolution of Generic Safety Issue 191 (CAC Nos. MF2400, MF2401, MF2402, MF2403, MF2404, MF2405, MF2406, MF2407, MF2408, and MF2409)," dated December 12, 2016 (ADAMS Accession No. ML16302A453).

22. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," June 1987 (Certain updated sections are available from the NRC).

23. U.S. Nuclear Regulatory Commission, NUREG-0800, Section 15.0.2, "Review of Transient and Accident Analysis Methods," December 2005 (ADAMS Accession No. ML053550265).

24. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.203, "Transient and Accident Analysis Methods," December 2005 (ADAMS Accession No. ML053500170).

25. Kaizer, J. S., Heller, A. K., and Oberkampf, W. L., "Scientific computer simulation review," Reliability Engineering and System Safety 138: 210-218, 2015.

26. Rencurrel, D. W., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "South Texas Project Units 1 and 2, Supplement 4 to the Response to Generic Letter 2004-02 (TAC Nos. MC4719 & MC4720)," dated December 11, 2008 (ADAMS Accession No. ML083520326).

27. Framatome ANP, EMF-2103(P)(A), Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," April 2003 (Not publicly available. Proprietary information). Non-proprietary version: EMF-2103(NP)(A), Revision 0, April 2003 (ADAMS Accession No. ML032691424).

28. U.S. Nuclear Regulatory Commission, NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980 (ADAMS Accession No. ML051400209).

29. U.S. Nuclear Regulatory Commission, NUREG/CR-5249, "Quantifying Reactor Safety Margins, Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident," December 1989 (ADAMS Accession No. ML030380473).

30. Powell, G. T., STP Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Description of Revised Risk-Informed Methodology and Responses to Round 2 Requests for Additional Information Regarding STP Risk-Informed GSI-191 Licensing Application," dated March 25, 2015 (ADAMS Accession No. ML15091A440).

31. H. C. No, K. W Lee, C. H. Song, "An Experimental Study of Air-water Countercurrent Flow Limitation in the Upper Plenum with a Multi-hole Plate," Nucl. Eng. Tech. 37, 557-564 (2005).

ENCLOSURE 4

RESOLUTION OF LICENSEE COMMENTS

ON NRC STAFF SAFETY EVALUATION

STP NUCLEAR OPERATING COMPANY, ET AL.

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

NRC Staff Resolution of STP Comments on Draft Safety Evaluation

NRC Staff Draft Safety Evaluation (SE) Editorial Comments from STP Nuclear Operating Company (STPNOC)

No.	SE or Attachment Section and Page	STPNOC Proposed Change	STPNOC Reason for Proposed Change	NRC Staff Resolution
1	Section 3.1, pg. 6	The sumps are located at the *Elev. (- minus)* 11-foot 3-inch level of the reactor containment building.	Correction for the proper floor elevation.	Change accepted.
2	Section 4.4.1, pg. 19	~~However, the licensee stated that piping in the containment is fabricated, designed, constructed, and examined (preservice inspections) with rigorous engineering requirements including safety factors.~~ However, the staff noted that the licensee described ASME code requirements for design, fabrication, construction, and examination of containment piping, and addressed associated safety factors.	Although we don't disagree and did describe ASME requirements, quality requirements, and safety factors, we could not identify where we specifically made this statement.	Change accepted.
3	Section 4.4.3, pg. 21	The licensee used guidance in RG 1.82 (remove note 51 re 2012 RG).	STP's UFSAR licensing basis is identified in the LAR as RG 1.82 draft Rev. 1, 1983.	Change accepted. Added new endnote 58.

No.	SE or Attachment Section and Page	STPNOC Proposed Change	STPNOC Reason for Proposed Change	NRC Staff Resolution
4	Section 4.5.2.2, pg. 28, 29	In its letter dated October 20, 2016, the licensee provided estimates of the risk attributable to debris. The licensee presented risk results using the arithmetic mean ~~in one case~~ and the geometric mean to allow comparison ~~in another~~. The licensee also stated its licensing position that the geometric mean is the most appropriate method and provided its basis in its response to APLAB, Results Interpretation – Uncertainty Analysis: RAI 2. Per NUREG-1829, providing analysis results under differing assumptions helps identify the sensitivity of the results to those assumptions. The NRC staff reviewed the licensee's information and concludes that the sensitivity analysis of the risk results to the choice of aggregation method is an acceptable way to address this source of uncertainty because it is consistent with the recommendation in NUREG-1829.	The SE phrasing suggests that STPNOC used geometric mean for one configuration and arithmetic mean for a different configuration and might apply the arithmetic aggregation for some conditions. Table 9 in Section 4.5.1 of Att. 1-3 to LAR Supplement 3 (10/20/2016) includes a "head to head" comparison of delta-CDF results for geometric and arithmetic means. STPNOC stated its licensing position in the paragraph below the table that the geometric mean is the most appropriate method and referenced its basis in a RAI response.	Partially accepted; the edits to the second sentence are accepted. The addition of the new sentence is not accepted because the licensee's position is not significant to the NRC staff's review.
5	Section 4.5.2.6.2, pg. 34	For DEGBs, D is equal to the inner diameter of the pipe and a spherical jet is assumed.	Agree – contradicts the square root of 2 discussion mentioned in No. 6 below (but this – just D – is the correct interpretation).	No change, since the current SE version in ADAMS correctly states this sentence.

No.	SE or Attachment Section and Page	STPNOC Proposed Change	STPNOC Reason for Proposed Change	NRC Staff Resolution
6	Section 4.5.2.6.2, pg. 35	In case of a full pipe break, the licensee ~~defined an equivalent break size based on $\sqrt{2}$ times the inner pipe diameter (i.e., the DEGB size is the diameter of a circular opening twice the area of the inner cross-section of the pipe).~~ used a spherical ZOI based on the material L/D using the applicable pipe ID.	See Comment 5.	Change accepted.
7	Section 4.5.2.6.2, pg. 36	In case of a full pipe break, the licensee uses a spherical ZOI based on the material L/D using the applicable pipe ID ~~defined an equivalent break size based on $\sqrt{2}$ times the inner pipe diameter (i.e., the DEGB size is the diameter of a circular opening twice the area of the inner cross-section of the pipe)~~.	Is correctly stated in Section 4.5.2.6.2, page 34. See Comment 5.	No change, since the current SE version in ADAMS correctly states this sentence.
8	Section 4.5.2.7, pg. 46	~~Debris settling is not credited for fine debris in the debris transport analyses~~98.5% of fine debris is transported to the RCB recirculation pool.	LAR (August 20): The majority of fiber fines (98.5%) destroyed from insulation in the ZOI are transported to the containment pool. The other 1.5% of debris not transported to the RCB sump is trapped in inactive cavities during pool fill. The transport modes and their contributing fractions to the containment pool for ZOI-generated fiber fines are described below.	Change accepted.

SE Attachment 2, "Long-Term Core Cooling Methodology and Evaluation Results Assessment" Comments from STPNOC

No.	SE or Attachment Section and Page	STPNOC Proposed Change	STPNOC Reason for Proposed Change	NRC Staff Resolution
1	LTCC Methodology, Section A.4.1, pg. 9	The licensee provided justification ... the licensee referenced responses to previous questions 31 and ~~35~~ <u>36</u>	Reference should be to Question 36 (See draft SE reference 27, pg. 74 of 77).	Change accepted.
2	LTCC Methodology, Section A.4.2.1.1, pg. 11	The licensee provided a description of the structured process used to identify and define the accident scenario in response to SNPB-3-2 (Reference 24 and 25) and SNPB-3-4 (Reference 22). ~~STPNOC stated that the process for accident scenario identification focused on three areas~~: <u>The staff review of the STPNOC process determined that it addressed the areas below</u>.	Could not find that we specifically made this statement, although the staff might have concluded from the review of the responses that STP put appropriate focus on these areas.	Change accepted.
3	LTCC Methodology, Section A.4.2.1.4, pg. 19	~~However, the licensee cautioned that the figure of merit for determining the limiting break should not be PCT, but the core collapsed liquid level~~.	We cannot find this cautionary statement. The collapsed liquid level is important, but PCT is the accepted regulatory figure of merit.	Partially accepted; modified as: However, the licensee *recognized* that the figure of merit for determining the limiting break should not be PCT, but the core collapsed liquid level.

No.	SE or Attachment Section and Page	STPNOC Proposed Change	STPNOC Reason for Proposed Change	NRC Staff Resolution
4	LTCC Methodology, Section A.4.2.2.5, pg. 23	Because the LTCC EM is only used to perform the simulations described in the RAI responses provided by the licensee, the NRC staff has determined that consideration of future licensing calculations was not needed. The NRC staff has concluded that this criterion does not apply. (No change suggested)	Check for understanding: If STP needs to rerun in the future (for the same cases but for, say a different block limit, more or less, different blockage timing, and so forth, we should be able to do so provided we use the same methodology that was reviewed.	Disagree. The licensee is limited in its use of this methodology as specified in the A.4.3 Conclusions.
5	LTCC Methodology, Section A.4.2.2.7, pg. 24	While the licensee provided justification for the use of the LTCC EM for the simulation of the 16-inch hot-leg breaks (and various sensitivity studies), complete accident-specific guidelines were not provided as the approval was limited to only those simulations already submitted to the NRC. Therefore, future use of the LTCC EM beyond the methodology and application reviewed by the staff requires prior review and approval by the NRC staff. Thus, the NRC staff concludes that this criterion does not apply.	We would understand this to still allow STP to apply the STP EM for similar sensitivity studies using the same methodology reviewed by NRC.	Disagree. The licensee is limited in its use of this methodology as specified in the A.4.3 Conclusions.
6	LTCC Methodology, Section A.4.2.3.4, pg. 27	The CCFL model is applied at the upper nozzles ~~core plate, a plate containing numerous holes which separates the fuel from the upper plenum~~.	Upper nozzles are where the CCFL is of concern rather than the upper core plate, per se.	Partially accepted. Modified to state "top of the core" versus "core plate." The NRC staff notes that CCFL is typically checked for both the top of the fuel (i.e., upper nozzles) and the upper core plate, and is applied at which ever one has the least flow area.

No.	SE or Attachment Section and Page	STPNOC Proposed Change	STPNOC Reason for Proposed Change	NRC Staff Resolution
7	LTCC Methodology, Section A.4.2.3.4, pg. 27	Wallis with smooth edges over-predicted the superficial velocities in data with a few number of holes, but under-predicted the superficial velocities in data with a moderate number of holes and greatly under-predicted the superficial velocities of data with a large number of holes. This was the correlation chosen by the licensee for the LTCC EM. (No change suggested)	Comment: In our understanding, CCFL is correlated by the superficial velocity of the steam (which is assumed positive), and superficial velocity of the liquid. When the liquid velocity equals the steam velocity, it is stopped (flow begins to be counter-current).	No changes made. The NRC staff notes that CCFL has to do with the velocities of the steam and the liquid, but it also has to do with the liquid/vapor interface and the friction created in each.

8	LTCC Methodology, Section A.4.2.3.4, pg. 28	Therefore, a likely SI flow path is down through the core support plate on the periphery, down the periphery fuel bundles, and then into the center of the core to make up for any loss due to boil off. Given the open lattice nature of the core, this flow path would in reality be almost unaffected by CCFL. However, due to the manner in which the core is meshed and the CCFL model is applied, this flow path is not possible in the simulation. The NRC staff finds that this is likely to be a conservatism in the licensee's analysis. (No change suggested)	This same argument should apply for cores that are not designed as "low leakage" since power sharing will always be an artifact of multi-region cores. For example, even with low leakage design, we showed that internal low power sharing regions had downflow.	Accepted and modified: Original: *In reality, while CCFL may occur in channels which have significant amounts of steam generated, such as center regions of the core, it would likely not occur near the core periphery. Therefore, a likely SI flow path is down through the core support plate on the periphery, down the periphery fuel bundles, and then into the center of the core to make up for any loss due to boil off. Given the open lattice nature of the core, this flow path seems almost unaffected by CCFL, but due to the manner in which the core is meshed and the CCFL model is applied, this flow path is not possible in the simulation. The NRC staff finds that this is likely to be a very large conservatism in the licensee analysis.* Modified: *In reality, while CCFL may occur in hotter channels which generate significant amounts of steam (e.g., those commonly found in central regions of the core), it would not be likely to occur in lower power channels (e.g., those commonly found in the core periphery). Therefore, a likely SI flow path is down through the core support plate on the periphery, down the periphery fuel bundles, and then into the center of the core to make up for any loss due to boil off. Given the open lattice nature of the core, this flow path would in reality be almost*

No.	SE or Attachment Section and Page	STPNOC Proposed Change	STPNOC Reason for Proposed Change	NRC Staff Resolution
				unaffected by CFL. However, due to the manner in which the core is meshed and the CCFL model is applied, this flow path is not possible in the simulation. The NRC staff finds that this is likely to be a conservatism in the licensee's analysis.
9	LTCC Methodology, Section A.4.2.4.2, pg. 32	... While CCFL was found to restrict the flow from the upper plenum into the core in the analysis, it is possible that this results from an oversensitivity to CCFL caused by modeling the core and thus the ~~upper core plate~~ *top nozzles* where CCFL occurs, as a single radial node.	CCFL will occur at the top nozzles, which were modeled in STP's LTCC EM.	Partially accepted; modified as: While CCFL was found to restrict the flow from the upper plenum into the core in the analysis, it is possible that this results from an oversensitivity to CCFL caused by modeling the core – *specifically, the top of the core where CCFL occurs* – as a single radial node.
10	LTCC Methodology Section A.4.2.4.7, pg. 35	<u>The staff believes</u> this is likely because in the two channel model, liquid can flow into the average channel and steam and liquid exit through the hot channel, but in the base case all liquid flowing into the core and all steam and liquid exiting the core must go through the same channel (i.e., the same node).	Should clarify that this is the staff's opinion. It is not included in our assessment and we have not performed analyses to support it.	Change accepted.

No.	SE or Attachment Section and Page	STPNOC Proposed Change	STPNOC Reason for Proposed Change	NRC Staff Resolution
11	LTCC Methodology, Section A.4.2.5, pg. 37	Safety parameters are those parameters which have limits to ensure plant safety, such as the specified acceptable fuel design limits (SAFDLs) required by General Design Criterion 10 from 10 CFR 50 Appendix A. Examples of safety parameters are PCT, cladding oxidation thickness, departure from nuclear boiling ratio (DNBR), and critical power ratio (CPR). No explicit uncertainty analysis was prescribed or performed for the LTCC EM. However, the NRC staff reviewed specific aspects of the LTCC EM to confirm that specific uncertainties would be accounted for in the analysis. (No change suggested)	Confirming the intent of the staff's statements: The uncertainties we addressed were pertaining to the debris issue in LTCC. It appears from the following paragraph that those were reviewed and found to be acceptable.	While STPNOC's comment did not request any change to the SE, they did request confirmation of the staff's intent. They clarified "that the uncertainties addressed were pertaining to the debris issues in the LTCC." While there were some uncertainties associated with the debris itself in the LTCC simulations, these were very limited. For example, the only direct uncertainty associated with debris was on the blockage time of the core inlet. This uncertainty was addressed by assuming a conservatively short blockage time. There were other uncertainties associated with the LTCC simulation which the staff considered to be the dominating uncertainties and which were addressed in the SE.
12	LTCC Methodology, Section A.4.3, pg. 46	The NRC staff's conclusions herein are specific to the South Texas Project and future uses of this LTCC EM require prior review and approval by the NRC staff for those specific details and plant design. (No changes suggested)	We understand this means other plants cannot use this without prior NRC approval; however, STP can continue to use it in accordance with the methodology we submitted.	Disagree. The licensee is limited in its use of this methodology as specified in the A.4.3 Conclusions.

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: CHANGES TO DESIGN BASIS ACCIDENT ANALYSIS USING A RISK-INFORMED METHODOLOGY TO ACCOUNT FOR DEBRIS IN CONTAINMENT (CAC NOS. MF2400 AND MF2401) DATED JULY 11, 2017

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